IRRADIATION EFFECTS STUDIES OF NERVA MATERIALS

J. A. DeMastry and T. P. Merrick

Westinghouse Astronuclear Laboratory

Several materials, Beryllium, CuB, BATH (Al-24 v/o TiH_2 - 6 v/o B_4C) and A-286, were irradiated at cryogenic temperatures to neutron fluences between 10^{17} and 10^{19} n/cm^2 (E > 1.0 Mev). Tensile properties were determined over a temperature range to evaluate the effects of irradiation on the material properties. The expected radiation damage typical of other materials was observed, a general increase in strength and decrease in ductility. The effects at LN_2 temperature (140°R) are more pronounced since radiation induced defects are not annealed out at 140°R.

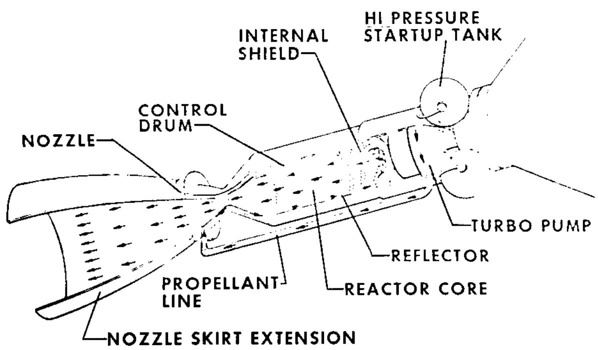
Decreases in both strength and ductility of Beryllium are noted at 140°R. Mechanical strength is generally recovered at temperatures as low as 406°R for 10 minute anneals. Thermal conductivity in beryllium is decreased by almost a factor of six at 1 x 10^{19} n/cm^2 (E > 1 Mev). Both Cu 12 v/o B and Cu 24 v/o B exhibit increases in strength and decreases in elongation due to irradiation. BATH shows an increase in mechanical strength after irradiation and testing at 140°R. The A-286 alloy undergoes decreased ultimate strength and increased yield strength under the conditions studied. There were no significant changes in elongation at the temperatures studied. Comparison of test data from different reactors and the statistical techniques employed in this program are also reviewed.

The NERVA (Nuclear Engine for Rocket Vehicle Applications) Program was initiated in 1961 after the Los Alamos Scientific Laboratory had successfully demonstrated the basic principles in nuclear rocketry. The program, under the direction of the Space Nuclear Systems Office of NASA and the AEC, is being performed by the Aerojet Nuclear Systems Company as the prime contractor and Westinghouse Electric Corporation as the principal subcontractor for the nuclear subsystem development.

The basic operation of a nuclear rocket system is as follows. A simplified sketch of a nuclear rocket engine attached to a flight tank is shown in Figure 1. The engine delivers approximately 75,000 lbs of thrust at a specific impulse of 825 lb_f-sec/lb_m. The reactor produces 1511Mw of power and the core consists of clusters of graphite fuel elements surrounded by a beryllium reflector. The pump, which is driven by a turbine, increases the pressure of the liquid hydrogen to 1300 psia and provides approximately 91 lb/sec through the pump discharge line to the nozzle

FIGURE 1

NERVA ENGINE



Note:
The Nuclear Engine for Rocket Vehicle Application (NERVA) Program is administered by the Space Nuclear Systems Office, a joint office of the U. S. Atomic Energy Commission and the National Aeronautics and Space Administration. Aerojet Nuclear Systems Company as prime contractor for the engine system and Westinghouse Electric Corporation as subcontractor for the nuclear subsystem, are developing a nuclear propulsion system for space applications.

inlet plenum. The liquid hydrogen then flows through the regeneratively cooled nozzle tubes into the reflector of the reactor. After passing through the reflector and removing the radiation-deposited energy, the hydrogen enters the shield region at the forward end of the dome. The purpose of the shield is to decrease the radiation levels on the engine parts. The hydrogen passes through the reactor-fueled section and is heated to above 4000°R when it enters the thrust chamber formed by the convergent section of the nozzle. The hot hydrogen from the thrust chamber is expanded and accelerated by the nozzle, thereby producing the required thrust.

The nuclear rocket engine derives its primary advantage over chemical rocket engines from its use of the hydrogen as a propellant which results in very high specific impulse. Specific impulse (I_{sp}), which is the ratio of thrust produced to propellant flow rate, is a prime measure of a rocket engine's performance since it relates directly to the amount of propellant which must be carried to perform a mission. Since specific impulse is a function of the inverse of the square root of the molecular weight of the propellant, hydrogen with a molecular weight of two, is an ideal propellant. All chemical rocket engines combine fuel and oxidizer with resulting higher molecular weights. Thus, the nuclear rocket engine developes a specific impulse approximately double that of the best chemical rocket engine.

The high reliability requirements for the NERVA engine dictate an extensive Radiation Effects Program. Each material and component must be evaluated over a range of temperatures, fluences, and other pertinent variables to determine if performance degradation occurs. This has involved irradiation of materials and components to 1×10^{19} n/cm^2 (E > 1 Mev) at liquid nitrogen temperatures and subsequent post irradiation testing at cryogenic temperatures.

Major recent effort has focused on candidate NERVA materials and these data are being reported today. Classification as ductile or brittle is made on the basis of preliminary screening in an environment which duplicates, as nearly as possible, the NERVA application. Applicable information from literature sources and tensile test results are used in determining if the material is to be classified as ductile or brittle for NERVA use. A material is defined as "brittle" if it demonstrates a sensitivity to non-inherent flaws in such a way that it fails below its yield strength. A material is classified as "ductile" if it is insensitive to flaws that may be present. The results of the above discussed screening tests dictate the requirements for additional testing such as fracture toughness, which are currently underway. The ductile material discussed below and the other ductile material utilized in NERVA exhibit increased or relatively unaltered strengths following irradiation. Thus, unirradiated design properties are utilized in determining the stress allowables. The brittle materials are statistically evaluated for changes in fracture strength following irradiation. Testing programs are currently underway for the brittle materials reviewed in this paper and only the beryllium irradiated fracture properties are available for discussion today.

Westinghouse has employed reliability design methodology which departs somewhat from classical design approaches. A probabilistic design approach is employed whereby the material properties used in analysis are those properties identified with a 99% probability at the 95% confidence level, presuming a Gaussian distribution and considering the effects of temperature uncertainties and other potential variables. Thus, experimental statistics are employed and whenever possible, full factorial test matrices are utilized. Ordinarily, the requirement is for eight independent observations in determining the mean value for each level of a primary variable and the variance about that mean is determined with at least 15 degrees of freedom. The data is analyzed using Bartlett's Test for homogeneity of variance and the standard analysis of variance.

In selecting materials for cryogenic applications, the most important criteria are strength, ductility, and toughness. Because of this, the body-centered cubic materials which exhibit a ductile-to-brittle transition temperature can be eliminated for purely cryogenic application since this transition temperature occurs above liquid-hydrogen temperature for most body-centered cubic materials. The materials generally used at cryogenic temperatures are the face-centered cubic materials such as the aluminum alloys, austentic stainless steel, and nickel-base alloys. In addition, certain hexagonal crystal structured materials such as beryllium, magnesium, and titanium have also been considered. The yield strength, ultimate strength, and notch strength of the face-centered-cubic materials is generally higher at cryogenic temperatures than it is at room temperatures. In most cases, the reduction of area of the fcc material is reduced by lower temperature; the total elongation may be increased, or decreased depending on the materials. Elongation at cryogenic temperature depends on grain size, degree of prior cold work, and temperature.

Theoretically, irradiations at cryogenic temperatures are expected to cause greater changes in mechanical properties than do irradiations at room temperature for the same level of fast fluence. This assumes that some of the vacancies, and interstitials produced by fast neutrons will be annealed out at room temperature; cryogenic irradiations do not produce significant annealing of fast neutron defects.

There is presently limited data available concerning cryogenic nuclear radiation effects in structural materials. Most of the data obtained to date has been developed for use in the NERVA Program. Prior studies have shown that the threshold for observable mechanical property damage is as low as 1×10^{17} n/cm^2 when irradiation temperatures are below 140°R. Since certain structural components of the NERVA system will be exposed to fluence levels of approximately 10^{20} n/cm^2 during the 10 hour lifetime, radiation damage to these components could be a serious problem.

IRRADIATION FACILITIES

Irradiation studies were conducted at the General Dynamics/Fort Worth Ground Test Reactor (GTR) which is a heterogeneous, highly enriched, thermal reactor utilizing water as neutron moderator and reflector, as radiation shielding, and as coolant. Figure 2 is a plan view of the facility and Figure 3 is a cutaway view of the irradiation test cell and the reactor tank. During operation, the

FIGURE 2. GTR RADIATION EFFECTS TESTING FACILITY

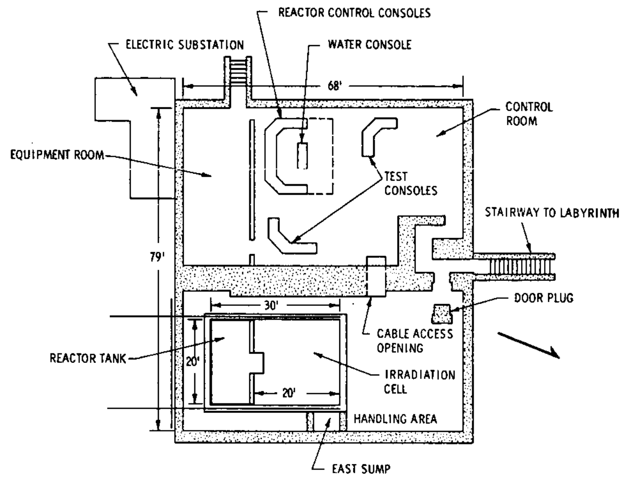
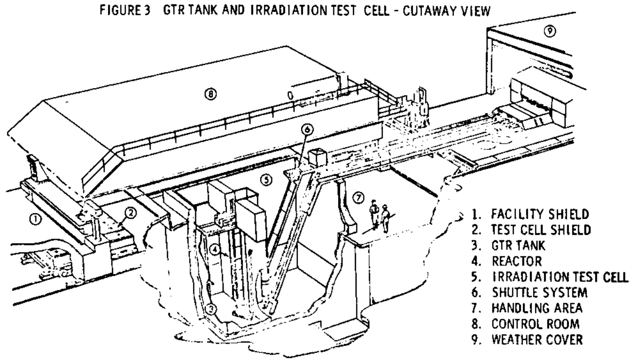


FIGURE 3 GTR TANK AND IRRADIATION TEST CELL - CUTAWAY VIEW



reactor is moved into the closet-like structure built into the north wall of the GTR tank. Specimens to be irradiated are located near the bottom of the test cell at the north, east, or west sides of the closet. Dewars containing the specimens to be irradiated are placed at the closet wall. The dewars are then filled with liquid nitrogen and a continuous level maintained in the dewar during the irradiation.

After irradiation, the dewars are removed from the test chamber to the General Dynamics/Fort Worth hot cells for post-irradiation examination and mechanical property studies. The liquid nitrogen level is maintained throughout transfer and storage. Specimens are never allowed to warm above liquid nitrogen temperature since such a warming would result in loss of damage due to annealing. The actual temperature of the specimens has been verified by locating thermocouples on selected specimens and monitoring the temperature throughout the irradiation, transfer, and storage phases.

Some data will also be discussed which are based on irradiation of beryllium thermal conductivity specimens in the Bulk Shielding Reactor (BSR), Oak Ridge National Laboratory (ORNL) at 200°R. Figure 4 depicts the lower end of the thermal conductivity test capsule whereby "in situ" measurements of thermal conductivity and electrical resistivity were performed. This has been intercalibrated with ORNL laboratory systems and good agreement observed.

The use of two test reactors described above, as well as the NASA Plum Brook Reactor at Lewis Research Center presents some interesting analytical problems. There are significant spectral differences between the NERVA reactor which is epithermal and these thermal water-moderated test reactors. For example, Jenkins and Williams at ORNL compared the differential flux spectra for the BSR and the NERVA radial leakage spectrum as shown in Figure 5 (normalized for E > 1 Mev). As a result, a correlation coefficient (based solely on beryllium displacement calculations) was developed. These results indicate that the reflector leakage spectrum is 1.4 times more damaging than the BSR position utilized if fluence is quoted for E > 1.0 Mev.

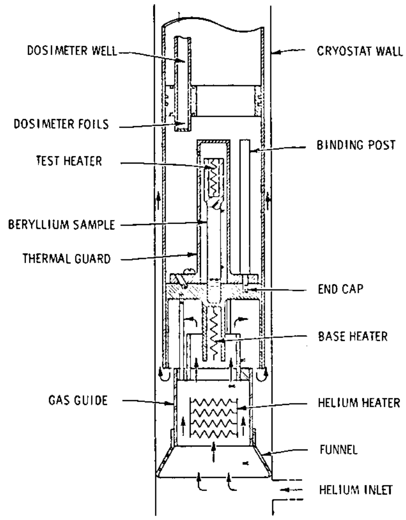


FIGURE 4.
ILLUSTRATION OF THE LOWER END OF THE THERMAL CONDUCTIVITY-TEST CAPSULE FOR THE BULK SHIELDING REACTOR LIQUID NITROGEN-COOLED CRYOSTAT. THE SAMPLE IS 2" LONG.

Westinghouse has performed similar machine and hand calculations for other materials and other reactors. The results indicate that a correction factor of from 1-3 is required in comparing a thermal-water-moderated reactor with NERVA.

POST IRRADIATION TESTING

Tensile tests were performed with a Model TT-D split-console Instron Tensile Test Machine having a variable-range load capacity of 20,000 lbs. Cryogenic testing was conducted in a cryostat constructed of urethane foam material; its inner and outer surfaces were coated with successive layers of siliconic adhesive and fiberglass cloth. All fracture mechanics tests were performed in the same facilities. Specimens were stored in LN_2 to prevent warming of specimens after irradiation with accompanying annealing of irradiation effects.

Beryllium thermal conductivity measurements reported today were all made "in situ" in the apparatus described above. The results are as follows.

FIGURE 5. COMPARISON OF DIFFERENTIAL NEUTRON FLUX SPECTRUM AT BSR EXPERIMENTAL LOCATION WITH THE NSS RADIAL LEAKAGE SPECTRUM

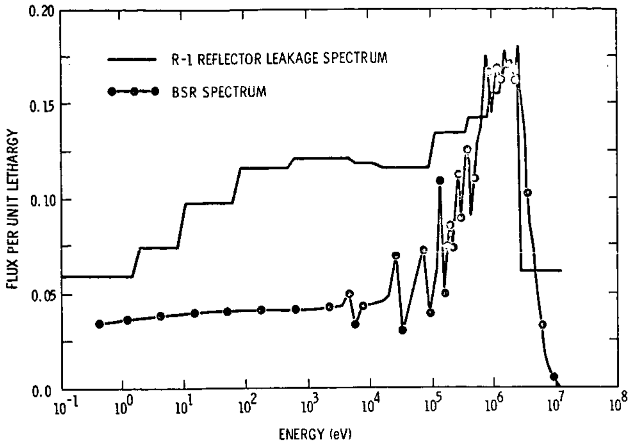


TABLE 1

TENSILE TEST DATA FOR BERYLLIUM IRRADIATED AT LN_2 (140^oR) TEMPERATURE TRANSVERSE DIRECTION

Test Temperature ^oR	Neutron Fluence 10^{18} n/cm^2 (E > 1.0 Mev)	Neutron Fluence 10^{18} n/cm^2 (E < 0.48 ev)	0.2% Offset Yield Strength KSI	Ultimate Tensile Strength KSI	Elongation %
140	0	0	38.2	52.3	1.51
	.46	.16	---	27.9	0
	2.4	.13	---	48.0	0
	7.5	.35	---	33.5	0
273	0	0	31.2	46.2	2.11
	.45	.16	---	50.0	0
	2.5	.13	---	48.6	0
	7.8	.35	---	49.1	0
406	0	0	32.3	49.2	2.52
	.43	.16	45.0	56.0	1.80
	2.5	.13	0	50.4	0
	8.3	.36	0	58.9	0
540	0	0	31.1	44.5	2.39
	.45	.16	30.6	43.7	2.27
	2.4	.13	34.4	40.2	1.05
	7.6	.35	33.1	45.0	2.12

TABLE 2

TENSILE TEST DATA FOR BERYLLIUM IRRADIATED AT LN_2 (140^oR) TEMPERATURE LONGITUDINAL DIRECTION

Temperature ^oR	Neutron Fluence 10^{18} n/cm^2 (E > 1.0 Mev)	Neutron Fluence 10^{18} n/cm^2 (E < 0.48 ev)	0.2% Offset Yield Strength KSI	Ultimate Tensile Strength KSI	Elongation %
140	0	0	33.7	37.8	0.53
	.46	.16	---	32.5	0
	2.4	.13	---	25.0	0
	7.7	.35	---	29	0
273	0	0	38.1	48.4	1.4
	2.4	.13	---	26.1	0
	7.8	.35	---	35.7	0
406	0	0	36.0	47.6	1.41
	.44	.16	43.8	49.9	0.73
	2.5	.13	---	50.5	0
	8.1	.36	---	45.2	0
540	0	0	33.1	42.1	1.45
	.43	.16	33.5	42.5	1.43
	2.4	.13	35.3	42.2	1.02
	7.5	.35	36.0	40.0	0.52

TEST RESULTS

Results of beryllium irradiation in liquid nitrogen and post-irradiation tensile testing are shown in Tables 1 and 2. Specimens tested at 140 and 273^oR show decreased elongation and ultimate tensile strength. All specimens fractured prior to yielding. Almost complete recovery of mechanical properties is noted for specimens tested at 540^oR for both transverse and longitudinal specimens. Recovery is also noted in material irradiated to 4.3×10^{17} n/cm^2 and tested at 406^oR. Figure 7 depicts typical changes in UTS as a function of fluence at both 140^oR and 540^oR.

Table 3 shows the results of annealing studies to establish the time required for recovery of mechanical properties. Recovery of ultimate strength is noted at temperatures as low as 273^oR after 1000 minutes. Complete recovery of ultimate strength is observed at 406^oR and above for annealing times as low as 10 minutes. Ductility measured as elongation is not recovered until temperatures as high as 674^oR are reached. Even at this temperature, prior irradiation ductility is not fully recovered.

TABLE 3

EFFECTS OF ANNEALING ON BERYLLIUM IRRADIATION AT LN_2 (140^oR) TEMPERATURE ALL TESTS CONDUCTED AT 140^oR

Annealing Time: Time (min)	Annealing Time: Temperature ^oR	Neutron Fluence 10^{18} n/cm^2 (E > 1.0 Mev)	0.2% Offset Yield Strength KSI	Ultimate Tensile Strength KSI	Elongation %
0	140	7.1	--- ---	31.8	0
10	273	7.7	---	19.7	0
100	273	7.9	---	24.5	0
1000	273	7.4	---	30.1	0
10	406	8.4	---	31.9	0
100	406	7.8	---	26.2	0
1000	406	7.6	---	44.8	0
10	540	7.6	---	32.5	0
100	540	8.1	---	31.6	0
1000	540	7.6	---	33.1	0
10	674	7.6	39.2	39.3	.01
100	674	7.7	---	35.2	0
1000	674	7.7	39.3	41.6	0.3

Figure 6 depicts the observed beryllium fracture toughness as a function of temperature for specimens irradiated to 3×10^{18} n/cm^2 and 8.5×10^{18} n/cm^2 (E > 1 Mev). As can be seen, irradiation significantly reduces the fracture toughness of beryllium. At 8.5×10^{18} n/cm^2, the fracture toughness (K_{1C}) is reduced by 30% at 140°R.

Other studies have indicated that the above damage can be annealed out by appropriate heat treatments. No annealing has been observed below 310°R and virtually complete annealing of damage is observed at 535°R.

The next figure (Figure 8) depicts the decrease in thermal conductivity in beryllium as a function of fluence. At 1×10^{19} n/cm^2 (E > 1 Mev), its thermal conductivity is reduced by almost a factor of six. This is completely recoverable at 400-500°R.

FIGURE 6. EFFECTS OF IRRADIATION ON FRACTURE TOUGHNESS

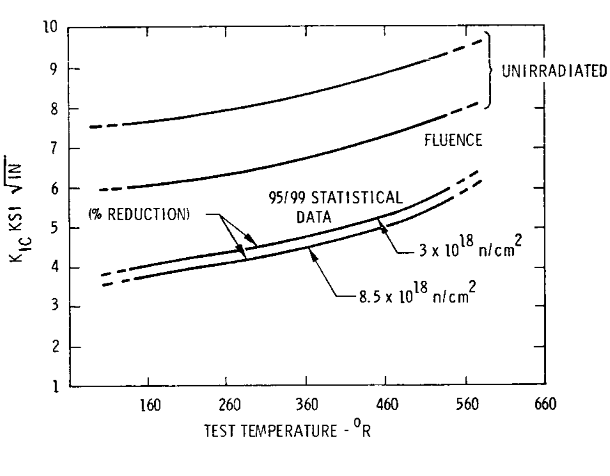


TABLE 4
TENSILE TEST DATA (1) for Cu 12 v/o B ALLOY IRRADIATED AT LN_2 (140°R) TEMPERATURE

Test Temperature °R	Neutron Fluence $1 \times 10^{18} n/cm^2$ (E > 1.0 Mev)	(E < .48 ev)	Orientation	0.2 % Offset Yield Strength KSI	Ultimate Tensile Strength KSI	Elongation %
140	0	0	L	11.5	38.0	30.4
140	0	0	L	11.8	34.1	21.7
140	2.31	0.12	L	46.2	48.8	2.74
140	2.38	0.12	L	48.4	49.3	4.15
540	0	0	L	9.4	24.1	18.4
	2.19	0.12	L	28.8	30.9	8.1
940	0	0	L	6.8	14.7	13.1
940	2.39	0.12	L	18.1	19.3	2.23
1340	0	0	L	3.8	5.6	2.59
1340	2.37	0.12	L	5.5	6.6	3.86

L - Longitudinal
(1) - Instron crosshead speed = 0.02 in/min

FIGURE 7. EFFECTS OF LN_2 (140°R) RADIATION ON BERYLLIUM TENSILE STRENGTH (LONGITUDINAL)

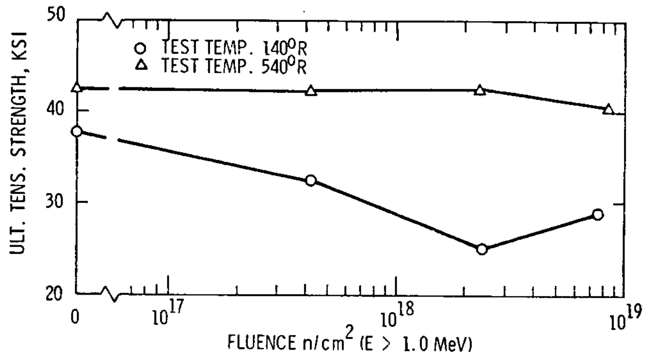
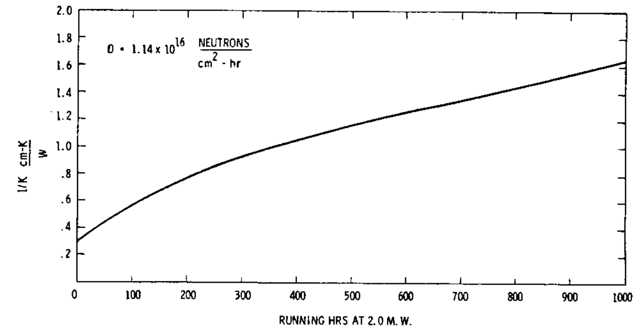


FIGURE 8. EFFECT OF RADIATION ON THERMAL CONDUCTIVITY OF BERYLLIUM



A copper-boron material is being considered for utilization as control drum vane material of the reflector assembly. Varying compositions of this alloy have been considered. Test results for two of these materials, Cu 12 v/o B and Cu 24 v/o B, are shown in Tables 4 and 5. The Cu 12 v/o B alloy exhibits large increases in yield and ultimate strength at temperatures up to 940^{o}R. These strength increases are accompanied by reductions in elongation. The Cu 24 v/o B alloy shows much smaller increases in strength in material tested at 540 and 940^{o}R with decreases in elongation. Material tested at 1340^{o}R showed no effects due to irradiation at 2.36×10^{18} n/cm^2.

TABLE 5
TENSILE TEST DATA [1] for Cu 24 v/o B ALLOY IRRADIATED AT LN_2 (140^{o}R) TEMPERATURE

Test Temperature ^{o}R	Neutron Fluence 1×10^{18} n/cm^2 (E> 1.0 Mev)	(E > .48 ev)	Orientation	0.2% offset Yield Strength KSI	Ultimate Tensile Strength KSI	Elongation %
140	0	0	L	11.3	27.3	11.8
140	2.29	0.12	L	8.3	14.8	1.79
540	0	0	L	7.2	19.5	2.1
540	2.15	0.12	L	13.3	24.7	0.08
940	0	0	L	8.0	12.5	6.0
940	2.32	0.12	L	13.3	14.1	1.64
1340	0	0	L	3.6	5.5	3.82
1340	2.36	0.12	L	3.4	5.4	4.04

L - Longitudinal
(1) - Instron crosshead speed = 0.02 in/min

The BATH (Al-24 v/o TiH_2 - 6 v/o B_4C) material is to be used as a neutron shield in the NERVA reactor. The results of irradiation at LN_2 (140^{o}R) temperatures and testing between 140 and 1140^{o}R are shown in Table 6. Material tested at 140^{o}R show increases in the ultimate strength and slight decreases in elongation. Testing at 340 to 1140^{o}R produces very little change in the mechanical strength of BATH. While increasing the fluence levels at any test temperature above 140^{o}R does not have a significant effect on the strength, it does produce changes in elongation. No pattern for this change can be noted. Both increases and decreases in elongation are noted. These changes are generally small and probably of no practical significance.

TABLE 6
TENSILE TEST DATA FOR BATH (Al-24 v/o Ti H_2-6 v/o B_4C) IRRADIATED AT LN_2 TEMPERATURE

Test Temperature ^{o}R	Neutron Fluence 10^{18} n/cm^2 (E >1 Mev)	0.2% Offset Yield Stress (KSI)	Ultimate Tensile Strength KSI	Elongation %
140	0	15.9	22.9	1.4
	2.78	-	28.9	.36
	3.4	-	34.4	.84
340	0	13.8	20.0	3.2
	2.81	20.5	22.1	1.3
	3.31	-	20.78	.72
540	0	11.9	18.8	4.89
	2.68	12.8	18.9	5.75
	3.45	12.8	18.5	2.86
740	0	11.3	14.7	7.39
	2.87	11.7	14.8	8.94
	3.21	11.4	14.8	6.55
940	0	8.9	10.6	5.36
	2.92	8.8	10.6	6.36
	3.14	9.0	10.2	8.24
1140	0	6.1	6.9	5.0
	3.04	6.1	7.1	4.3

TABLE 7
TENSILE TEST DATA FOR A-286 ALLOY IRRADIATED AT LN_2 (140^{o}R) TEMPERATURE

Test Temperature ^{o}R	Neutron Fluence 10^{18} n/cm^2 (E > 1.0 Mev)	0.2% Offset Yield Strength KSI	Ultimate Tensile Strength KSI	Elongation %
140	0	112.0	197.3	43.3
	0.41	123.5	198.2	40.3
	2.56	147.8	201.7	37.4
	10.5	177.5	200.6	34.9
340	0	101.2	163.2	26.1
	0.41	112.6	163.7	29.6
	2.56	132.5	159.3	28.0
	10.5	139.4	153.1	22.5
540	0	96.7	152.1	26.0
	0.41	106.5	150.0	28.3
	2.56	123.3	147.0	27.0
	10.4	120.0	136.5	22.0
740	0	90.1	144.9	23.7
	0.41	101.0	143.3	24.1
	2.56	112.4	141.3	20.6
	10.4	115.2	131.2	18.5
940	0	81.2	138.5	18.9
	0.41	82.0	140.2	23.0
	2.56	88.9	132.9	21.7
	10.1	100.6	131.4	16.7

The A-286 stainless steel alloy is a prime support material for the NERVA reactor and these data are tabulated in Table 7. Irradiation at 140°R with testing from 140 to 940°R produces only slight changes in the properties of A-286 as can be seen in Figure 9. Generally, increases in the yield strength due to irradiation are noted. Increases in yield strength due to decreasing temperatures are also observable. Irradiation also causes reductions in ultimate tensile strength at most test temperatures. Slight decreases in elongation due to irradiation are noted. The reductions in elongation are not large enough to be considered a serious problem.

CONCLUSIONS

1. Irradiation to about 8.0×10^{18} n/cm^2 (E> 1.0 Mev) produces decreases in both strength and ductility of Beryllium at 140 and 273°R. The mechanical strength is generally recovered at 406°R for anneals as short as 10 minutes.
2. Beryllium thermal conductivity, at 200°R, is reduced by almost a factor of six at 1×10^{19} n/cm^2 (E>1Mev) and this is equivalent to approximately 7×10^{18} n/cm^2 (E>1 Mev) in the NERVA reflector.
3. Both Cu 12 v/o and Cu 24 v/o B show increases in strength and decreases in elongation due to irradiation.
4. BATH material shows increases in mechanical strength after irradiation and testing at 140°R. Reductions in elongation are noted. Irradiation at 140°R with testing at 340°R and above produces little effects on the properties of BATH up to fluence levels of about 3.4×10^{18} n/cm^2 (E> 1.0 Mev).
5. The A-286 alloy exhibits increased yield strength and generally decreased ultimate strength over the temperature range studied. Changes in elongation were insignificant.

FIGURE 9. EFFECT OF LN_2 (140°R) RADIATION ON A-286 YIELD STRENGTH