Effects of Neutron Radiation (or Radiation Damage) on Niobium and Niobium Alloys: A Bibliography

Compiled by
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EFFECTS OF NEUTRON RADIATION (OR RADIATION DAMAGE) ON NIOBIUM AND NIOBIUM ALLOYS:
A BIBLIOGRAPHY

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Nuclear Science Abstracts, v.22-23 (1968-69) was searched for information on effects of neutron radiation (or radiation damage) on niobium and its alloys. Entries are arranged in the order they appeared in Nuclear Science Abstracts.
Brittle transition temperature of all the polycrystalline material is used. The results indicate that point defects in the crystal lattice increased by other work, and the present experimental results are discussed in terms of these loops. (auth)

Etch-pit and electron microscopy studies in niobium are described. A theoretical investigation of slip on the b.c.c. (110) and (112) planes is also given. (TAB)

The superconducting magnetization behavior and transition temperatures of single crystals of Nb were investigated prior to and after a series of fast-neutron irradiations (E > 1 MeV) in the Oak Ridge Research Reactor at a temperature of approximately 40°C. The experimental results indicate that the point defects and defect aggregates which are created lead to increases in the upper critical field Hc2 and small changes in the superconducting transition temperature. In addition, the increase in nonequilibrium behavior of the magnetization with fast-neutron flux establishes that defects are present with the ability to pin magnetic flux. For fast-neutron doses of approximately 2 x 1018 neutrons/cm2, the magnetization approaches a behavior that of a completely irreversible type-II superconductor. Successive reductions in cross-sectional area of the sample by chemical etching and subsequent measurements of the magnetization indicate only a small size dependence of the hysteresis, indicating essentially complete flux pinning in a macroscopic sheet approximately 0.1 mm thick. The flux-maintaining defects are considered to be dislocation loops as evidenced by other work, and the present experimental results are discussed in terms of these loops. (auth)

The radiation embrittlement of weld heat-affected-zone (HAZ) samples and base-plate samples of ASTM A-212B pressure vessel steel is being investigated as a function of radiation dose and temperature. The increase in ductile-brittle transition temperature is sharply reduced for irradiation temperatures above 300°C. Preliminary results indicate no systematic differences in dependence on irradiation temperature for HAZ samples as compared with base-plate samples. Samples shielded with cadmium exhibited the same degree of embrittlement as unshielded ones, thus suggesting that the embrittlement is relatively insensitive to thermal neutrons. Quantitative measurements of the relative amounts of shear and cleavage fracture appearance were made; the fracture appearance indicates a narrower transition from ductile to brittle behavior than the fracture energy, but the two properties are in essential agreement. Tensile tests, electrical resistivity measurements, and internal friction measurements were conducted on unirradiated and irradiated vacuum-melted Fe with and without the addition of 20 wt ppm of N. Annealing experiments following irradiation at ~30 to ~95°C showed an annealing stage at 35°C in the Fe - 20 wt ppm N material that is not present for the N-free Fe. This is attributed to the trapping of N at radiation-produced defects. The tendency to yield stress upon irradiation, followed by a further increase upon annealing in the range ~33 to 100°C. However, this was observed for samples of both N contents. Defect clusters and dislocation channeling were observed in neutron-irradiated Nb for the first time; a dose of 2 x 1019 neutrons/cm2 (E > 1 MeV) produced a spot density of approximately 5 x 1010 per cm2. The spot size distribution showed a peak at a spot diameter of about 30 A. As a result of resistivity and internal friction measurements, the post-irradiation annealing stage in Nb at 150°C ("Stage III" annealing) was shown to be due to the motion of O to radiation-produced defects. The activation energy was 10 eV. No significant annealing was observed after cold work for a low-O Nb sample. Temperature dependence of the yield stress in single-crystal Nb was virtually unchanged by neutron irradiation, in agreement with previous results from Fe indicating that the mechanism for the thermal activation of flow is not affected by the radiation-produced defects. A mechanism of radiation hardening in bcc metals by contact interaction between dislocations and damage clusters is formulated and is compared with the experimental results on irradiated Nb. The stress dependence of dislocation velocity was measured directly by etching techniques for unirradiated Nb as a function of temperature and number of zone-passes and for irradiated single-zone-pass Nb at room temperature. The equation \( V = (\pi/2) \phi \) was obeyed. For decreasing test temperatures in the range 300°C to 77K, m2 is not greatly changed and \( \tau_0 \) is greatly increased. Preliminary results at room temperature indicate a decrease in \( m^* \) and an increase in \( \tau_0 \) upon irradiation. Indirect measurements of \( m^* \) by change-in-strain-rate tests and stress-relaxation tests did not agree with \( m^* \) determined by the etching technique. (auth)

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The effect of neutron irradiation on the mechanical properties of Nb and its alloys depends on their purity. Difference in the hardness of the irradiated materials is probably connected with the unequal degree of structure defects in the original state. The neutron current at the energy given leads to the formation of a definite quantity of defects. If a material contains a large number of defects, then the neutron irradiation does not lead to the formation of new defects and strengthening does not occur. Neutron irradiation leads to changes in the antifriction friction processes as a result of the increase of the number of defects in its structure. (tr-auth)


An investigation of the annealing of defects in niobium and tantalum, induced by electron-irradiation using 1.95 MeV electron beam is described. The specimens were irradiated at 13.2°K and subsequently annealed at temperatures up to 301°K. Isothermal-annealing procedures were used. Tantalum specimens showed six main recovery peaks at temperatures 27.4, 34.2, 40.8, 53.0, 124.3, and 157.8°K with associated activation energies ranging from 0.605 to 0.646 eV and no observed recovery peaks from 135°K to 301°K. Niobium specimens showed seven recovery peaks at temperatures 14.9, 17.8, 24.6, 28.2, 55.8, 99.1, and 135.7°K with associated activation energies ranging from 0.043 to 0.392 eV and no observed recovery peaks from 135.7°K to 301°K. A brief sketch of the mathematical analysis for computation of these activation energies (in both liquid helium and liquid nitrogen temperature ranges) is given. Detailed comparisons are made between the damage production data, annealing spectrum for both metals and for those for noble metals. A comparison of the results with those obtained from neutron irradiation on niobium and tantalum is given. (Disser. Abstr.)

13036 (BNWL-473, pp 7.1-71) IRRADIATION DAMAGE TO REACTOR METALS. Bement, A. L. ( Battelle-Northwest, Richland, Wash., Pacific Northwest Lab.).

Inconel 600 has been irradiated to a fast fluence of \(1.37 \times 10^{14} \text{n/cm}^2\) (E > 1 MeV). Results of tensile tests show little difference in properties from those of specimens exposed to lower fluence levels. A total of 42 specimens was discharged from the ETR. Tensile data have been recorded, processed, and stored in the REM computer program for 5140 individual tensile tests. AISI Types 304 SS, 348 SS, 410 SS, and AM 350 SS were irradiated in the ETR hot water loop at 280°C. Results from tensile tests at 135°F on Inconel 600 and Hastelloy X 280 specimens held as controls for those irradiated at 1350°F to a fast fluence of \(1 \times 10^{13} \text{n/cm}^2\) (E > 1 MeV) and metallographic examinations indicate that thermal exposure at this temperature causes significant hardening and ductility loss as a result of precipitation reactions. Preliminary data from stress-rapture tests of Inconel X-750 at 1330°F indicate that irradiation at 43°F to a fast fluence of \(1 \times 10^{13} \text{n/cm}^2\) decreases rupture life by 65%. In-reactor creep rupture test results on AISI 304 SS have indicated a much greater effect of irradiation on rupture life at 840°C than at lower temperatures. A faster second-stage creep rate has been observed during irradiation than in an uniradiated control test of heat-treated 2-1/2 wt.\% Nb at a stress of 25.6 kg/mm² and 300°C. Uniaxial creep tests performed on longitudinal sections of Zircaloy-2 pressure tube material at 31.5 kg/mm² stress and 300°C have resulted in a faster in-reactor creep rate than the corresponding creep rate for the unirradiated control specimen. The fracture toughness of Zircaloy-2 has been determined after cold work, hydrogen charging, irradiation, and on-out-of-reactor specimens. Annealing irradiated, cold-worked Zircaloy-2 at 475°C produced a significant increase in toughness in the first 50 minutes. A study was begun to determine the size effects on the fracture toughness of low-strength, high-toughness steels using the double cantilever beam 0.5 in. and 1 in. thick DCB specimens and tests made on a quenched and tempered 4 in. thick plate of AISI-82-2 steel. The high pressure forms of Zr, Ce, InTe, Cds, GdC₂, and BeC₂ were produced and x-rayed. The high pressure polymorphs of Zr, InTe, and GdC₂ were quenched to ambient conditions. The cube-monoclinic transformation in GdC₂ is reported and the first known x-rays of crystalline banezene at 22°C and 11 kbars were obtained. Zircaloy-2 tensile specimens were examined after G-7 loop exposures. Sixty tubular Zircaloy-2 and Zircaloy-4 corrosion specimens were exposed 208 days in the Plutonium Recycle Reactor. Corrosion studies in the ATR model gas loop demonstrate that molybdenum and tungsten do not undergo chemical reaction with the impurities in helium gas stream. Haynes 25 and Hastelloy X lose weight by an evaporation mechanism in some test locations and oxidize in other locations. Irradiation plus containing 50 specimens of AISI 304 and 348 SS in the annealed and cold-worked condition have been discharged from the ETR-II. Postirradiation thin film transmission electron microscopy of annealed 304 SS irradiated in the ETR-II has shown that the damage state consists of Frank loops and polyhedral voids. Liquid sodium capsule GEN 224-continues operation at peak sodium temperature at 1460°F. The building which will house the testing facilities is in design stage and progress in all areas is reported. (J.C.W.)


In nuclear reactors the structural materials are subjected more or less to radiations. They must not disturb the neutron flux and must retain their mechanical properties as long as possible. The metallurgical industry faces a hard task in adapting known materials and developing new materials to combine these requirements. The materials and alloys developed in the past years and used for nuclear technology are reviewed. (auth)


Creep-rapture tests at 650°C on Hastelloy X have shown the same reduction of properties regardless of whether the specimens were irradiated in the ETR or EBR-II to fluences of 0.5 to 3 \(\times 10^{15} \text{n/cm}^2\) (E > 1 MeV). Incoloy 800, irradiated to \(3 \times 10^{15} \text{n/cm}^2\) (E > 1 MeV) and tested at 750°C, shows about a factor of two decrease in minimum creep rate compared to unirradiated material. This change in creep rate is accompanied by a factor of 2 decrease in elongation at fracture and a factor of 2 increase in rupture life. Hastelloy R-235 which had been re-solutioned and re-aged following irradiation in the ETR exhibited about one-half the as-irradiated rupture life at 857°C. Analysis of creep data for A-286, containing two different B levels and irradiated to various thermal neutron fluences, shows that the reduction in the minimum creep rate is directly related to the formation of the heavy compound 287B. The increase in stress dependence n. (\(n \approx 0\)) appears to be independent of He concentration, however, and is constant over the range of irradiation conditions examined. Transmission electron microscopy observations of irradiated Hastelloy R-235 reveal the presence of circular halos of dislocations surrounding selected electrode particles. These duplex halos consist of an outer halo with an approximate radius of 3 microns and a central halo of 1 to 1.5 microns radius. These distances are approximately equal to the A particle and Li atom
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reciprocal distances in Ni. The effect of post-irradiation annealing on the 700°C creep-rupture properties of irradiated Mo (1 x 10¹⁸ n/cm², E₀ = 1 MeV) has been continued on a series of specimens containing a factor of eight greater C (~200 µm²) content than the previously reported series. Results confirm that the annealing temperature of greater than 900°C is necessary to remove the incubation period for delayed creep, and that a 1200°C anneal, although eliminating the incubation period, does not restore the properties to control values. The data suggest an effect similar to thermal hardening, with the formation of stable defects at temperatures between 900°C and 1200°C. Initial results of the effects of post-irradiation annealing on the room- and elevated-temperature properties of irradiated Nb sheet (2.1 x 10¹⁸ n/cm², E₀ = 1 MeV) show no indication of thermal hardening. A 1-hour anneal at 1000°C was required for complete recovery to unirradiation values, compared with a 600°C anneal reported previously on material irradiated to a lower flux. Hot-microhardness data were determined for Mo–0.5Ti, Ta, Ta–10W, and 306 alloy (W–300, 30Mg, at. %) hardness and 306 alloy hardness showed the 256 alloy to be harder in the 850°C to 1300°C temperature range, a result consistent with the modulus difference between W and Fe in the two temperature ranges. A comparison was made of the Mo–0.5Ti data with previous Mo and Mo–TfM data. Radiochemical analysis of ninety-three Fe, Ni, and Ti dosimeters has shown the preliminary fast-flux mapping of SBR tens 2 and 7 positions. Good agreement was obtained between the three types of dosimeters used in the investigation. (auth)


Radiation damage and radiation hardening and embrittlement in the body-centered cubic metals are discussed. The problem of post-irradiation hardening in pressure vessel steels and the applied aspects of the problem are under investigation. The research covers work on commercial steels, iron and iron alloys, and refractory metals. The significance of the research and its applicability to materials problems in nuclear technology are discussed. (auth)

17072 (AECL-7275) EFFECT OF WATER CHEMISTRY ON THE OXIDATION OF ZIRCONIUM ALLOYS UNDER REACTOR RADIATION. Lenart, J. E.; Bryant, P. R. C. (Atomic Energy of Canada Ltd., Chalk River (Ontario), Chalk River Nuclear Labs.), Feb. 1968. 16p., Dep. CFSTI, CAN $0.50.

With pressurized water coolant, reactor radiation was reported to have no effect on the high-temperature oxidation of Zircaloy-2. In contrast, enhanced oxidation was reported from reactors where the water is allowed to boil. Corrosion information is presented for two zirconium alloys, Zircaloy-2 and Zr–2.5% Nb, oxidized in reactor radiation by water containing excess oxidizing agents. Both oxidizing chemistry and damaging radiation (e.g. fast neutrons) are required to enhance the oxidation of Zircaloy-2; oxygen in the absence of neutrons has little or no effect, and neutrons do not result in enhanced oxidation under reducing chemistry conditions. The sensitivity of Zr–2.5% Nb alloy to radiation and water chemistry depends on its metallurgical structure but, in general, it is sensitive to oxygen under all environments. Radiation appears to have no accelerating effect on Zr–2.5% Nb oxidation than other that associated with oxygen production from the radiolysis of water. Addition of ammonia to the water being irradiated prevents the appearance of molecular oxygen in-reactor and reduces the oxidation of both Zircaloy-2 and Zr–2.5% Nb. (auth)


From Industry Meeting on Water Reactor Fuel Element Technology, Germantown, Md.

In-reactor creep tests on two zirconium alloys, Zircaloy-2 and Zr–2.5 Nb, were conducted to determine the effect of concurrent irradiation on creep at different stresses and temperatures. The design of the creep test capsule is described, and the tests results are presented. The Zr–2.5 Nb offers an advantage in low creep rates at high stress levels greater than 21 kg/cm² (30,000 psi). At lower stress levels both alloys have approximately the same in-reactor creep rates. (L.C.L.)


From Industry Meeting on Water Reactor Fuel Element Technology, Germantown, Md.

A series of in-reactor experiments was run to determine the influence of variables such as coolant composition, alloy composition, and pretreatment procedures, on the flux-induced corrosion of zirconium alloys. Six zirconium alloys were tested in the ETR G-7 loop operated with pH=10 LiOH and with pH=10 NH₄OH. Specimens were either as-etched or passivated. The results showed that all three variables are significant, and that the response of the different alloys to in-flux corrosion is complicated. Niobium appeared to increase resistance to in-flux corrosion in the alloys tested. (L.C.L.)


A survey is given of irradiation-effects studies made and in progress on several different materials under varying conditions. Tension specimens of Zircaloy-2 with varying hydrogen contents were irradiated to neutron doses above 7 x 10¹⁸ n/cm² (>1 MeV) at 40°C and to 5 x 10¹⁸ n/cm² (>1 MeV) at 220 to 270°C. Stress-relaxation tests of Zircaloy-2 and Zr–2.5% Nb are in progress at 270°C to 10¹⁸ n/cm² (>1 MeV). Tension specimens of annealed and cold-worked steels similar to AISI Types 304 and 316 were irradiated to 4 x 10¹⁷ n/cm² (>1 MeV) at 40°C and about 5 x 10¹⁷ n/cm² (>1 MeV) at 270°C. Tension specimens of nickel-base alloys (Hastelloy X, Incon 800, 304Cr/304Ni/Nb) were irradiated at 550 to 700°C to 10¹⁸ n/cm² (>1 MeV). Impact tests were made on AISI 403 steel (13% Cr, 4% Si) after 10¹⁷ n/cm² (>1 MeV) at 550°C. Work in progress on postirradiation creep, in-pile creep, in-pile fatigue, and in-pile corrosion is briefly described. The significance of irradiation-effects data is discussed for in-pile loop tubes in a materials testing reactor. (auth)


Problems concerning the mechanical properties of Zr and Zr alloys used in pressure tubes under reactor operating condition at elevated temperatures are discussed. The effects of hydrides and C or N alloy additives are considered. (F.S.)
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After a review of the corrosion behavior and mechanical properties of Zr-NbSn alloys, the in-pile behavior is discussed. Ten--
sile specimens and samples under corrosion attack were irradiated in the swimming pool reactor FRG-1. The specimens were examined metallographically, tested by impact and tensile tests, and the effect of a water change on corrosion behavior was investigated. (F.S.)


The known effects of neutron irradiation on the physical and mechanical properties of bcc metals and alloys are reviewed, with particular emphasis on measurements made at room temperature and above. Specifically, the effects of irradiation on resistivity, stored energy, lattice parameter, linear expansion, hardness, creep rupture, and tensile properties are presented. Changes in the brittle-to-ductile transition temperature are also discussed. Recent calculations of irradiation hardening in ferritic steels are compared with experimental results. Transmission electron microscope studies of the substructure of irradiated bcc metals are described. Activation energies for the migration of irradiation-induced defects in bcc metals are compared with those reported for the fcc metals. (auth)


The results of in-pile and post-irradiation tests of an experimental assembly of the icebreaker-type rod fuel element with the sintered UO2 and Zr- 1 3% Nb alloy cladding are presented. The results of metallographic and mechanical investigations are also presented. The assembly operated for 10000 hours in hot pressurized water without failure. (auth)

23841 (BARC-300) RADIATION DAMAGE STUDIES ON Zr-0.5 PERCENT Nb-1 PERCENT Cr ALLOY. Ray, S.; Sharma, B. D. (Bhabha Atomic Research Centre, Bombay (India)). 1967. 24p. Dep.

Reported is the effect of neutron irradiation up to an integrated dose of 1.6 x 1017 n/cm2 (thermal) on the tensile properties of Zr- 0.5% Nb-1% Cr in annealed, cold-worked and as-quenched states. The irradiation was carried out in Apsara at 50°C. The studies reveal that whereas the cold-worked and as-quenched alloy does not exhibit any significant hardening, in the annealed states ultimate tensile strength and 0.2% yield strength increase by 100% and 66% respectively for a dose of 1.25 x 1017 nvt. Isochronal and isothermal recovery studies further reveal that the irradiation-induced damage anneals out at 175 to 410°C with an activation energy of 1.3 eV. The temperature and neutron flux dependence of 0.2% yield strength of irradiated alloy shows that hardening could be attributed to irradiation-induced depleted zones, as suggested by Senger. (auth)


The stress-dependence of dislocation velocity was measured by etching in niobium single crystals grown by an electron-beam floating-zone technique. The velocity-stress relaxation was determined with temperature, purity, and low neutron dose levels as variables. The data can be expressed in all cases by an equation of the form \( \dot{\gamma} = (\dot{\tau}/\tau_0)^m \) where \( \dot{\gamma} \) is the average velocity, \( \dot{\tau} \) is the effective shear stress, and \( \tau_0 \) and \( m \) are constants. The exponent \( m \) was found to increase slightly with decreasing temperature, increase strongly with greater impurity content, and to decrease with neutron irradiation up to 3.3 x 1016 n/cm2 (E > 1 MeV). Measurements of the strain-rate sensitivity and the stress relaxation indicated that these techniques were unsatisfactory in determining the exponent \( m \) in the velocity equation. Correlations could be made between the data from etching and mechanical measurements in irradiated samples which suggest that the increase in the upper yield stress upon neutron irradiation is associated with a dynamic resistance to dislocation motion. (auth)


Results of tensile tests on heat-treated Zr-2.5 at. % Nb alloys, under constant creep loads both unirradiated and during neutron irradiation are reported. At 300 and 320°C the difference between the in-reactor creep rate and the unirradiated control creep rate shows little if any dependence on stress or temperature. At stress levels between 11 and 14 kg/mm2 the in-reactor creep rate of Zircaloys-2 is essentially the same as Zr-2.5 at. % Nb at stress levels from 25.6 kg/mm2 or lower. It is concluded that only at stress levels greater than approximately 26 kg/m2m2 does the Zr-2.5 at. % Nb alloy offer an advantage over Zircaloy-2. The rate-limiting or controlling step in the creep process is most likely the generation of point defects. (P.G.C.)


An irradiation-enhanced strain relaxation was observed in Zircaloy-4 and Zr/2.5 wt % Nb/0.5 wt % Cu for fast neutron fluxes of 2.4 x 1017 nvt > 1 MeV. The effects of several material and test parameters were investigated and the results compared with out-of-pile stress relaxation data using a thermally-activated creep model. It was found that in-reactor relaxation of Zircaloy-4 was temperature and structure sensitive, independent of initial stress, and consistent with an accelerated thermal creep model. (auth)
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To assist in the design of the pressure tube as a core component, the creep behavior of cold-worked Zircaloy-2, cold-worked zirconium-2.5 wt % niobium, and heat-treated zirconium-2.5 wt % niobium materials is studied by measuring the inside diameter of reactor pressure tubes during reactor shutdown. The results show a significant increase in creep rate which can be directly related to the fast neutron flux. The equipment and procedures used to measure diametral creep, and the results gained from several experimental pressure tubes in the NRU and NPD reactors, are discussed. The diametral creep rate of a cold-worked Zircaloy-2 pressure tube operating at 270°C, 3 × 10^10 n/cm^2·s (>1 MeV) and 14,000 psi under a transverse to longitudinal stress ratio of 2:1 (closed end tube) is about 1.8 × 10^-7/μ, and is about ten times the creep rate out-of-reactor. The effects of stress and temperature are not as well defined as the effect of flux. Results indicate that for tubes operating between 10,000 to 14,000 psi stress, the exponent n in the relation \( \dot{\epsilon} = A \sigma^n \) (strain rate, \( \sigma \) = stress) is closer to 1 than to the value of about 3 found in the in-reactor uniaxial tests. Results also indicate that the in-reactor creep rate under biaxial stressing is less temperature dependent than under uniaxial stressing. As a first approximation, the equation, creep rate \( \dot{\epsilon} = 4 \times 10^{18} \sigma^{1.9} \) (1-T - 160°C)/hr could be used for cold-worked Zircaloy-2 in the temperature (T) range from 250 to 300°C, hoop stress (\( \sigma_h \)) up to 20,000 psi, and with fast neutron flux (\( \phi \)) up to 3.5 × 10^{15} n/cm^2·s (>1 MeV). At present both uniaxially stressed specimens and tube results show the effect of fast flux but there is no direct correlation between them. The creep of cold-worked zirconium-2.5 wt % niobium is accelerated by the fast neutron flux but the rates are about one-third those of cold-worked Zircaloy-2. Results have been obtained at stresses from 15,500 to 23,000 psi at 270 to 285°C and up to 3.1 × 10^{15} n/cm^2·s fast neutron flux (>1 MeV). (auth)


Experimental results of 24 uniaxial in-reactor creep tests on various zirconium alloys, covering a range of stresses from 7 to 42 kg/cm^2 and at 200 to 400°C are shown. At 300°C, cold-worked zirconium-2.5 wt % niobium and heat-treated zirconium-2.5 wt % niobium and heat-treatment Zircaloy-2. Results have been obtained at stresses from 15,500 to 23,000 psi at 270 to 285°C and up to 3.1 × 10^{15} n/cm^2·s fast neutron flux (>1 MeV). (auth)

31419 (BNWL-532, pp 11L-43) RADIATION METALLURGY. Billington, D. S.; Wechsler, M. S.; Stanley, J. T. (Oak Ridge National Lab., Tenn.).

The radiation embrittlement of weld heat-affected-zone (HAZ) samples and base plate samples of ASTM A-212B pressure vessel steel is described as a function of radiation dose and temperature. Tensile tests, electrical resistivity measurements, and internal friction measurements are presented for unirradiated and irradiated vacuum-melted Fe with and without 20 wt ppm of nitrogen. The temperature dependence of the yield stress in single crystal Nb is unchanged by neutron irradiation. The stress dependence of dislocation velocity was measured directly by etching techniques for unirradiated Nb as a function of temperature and number of zone passes and for irradiated single-zone-pass Nb at room temperature. 95 references included. (auth)

31421 (ORNL-4246, pp 1-8) RADIATION METALLURGY, Billington, D. S.; Wechsler, M. S. (Oak Ridge National Lab., Tenn.).

A review of research objectives is presented. It is noted that the principal metals under investigation are pressure vessel steels, Fe, Fe alloys, Nb, and V. Most effort is being devoted to mechanisms of radiation embrittlement, and the annealing characteristics of irradiated steel. (J.R.D.)


Internal friction—temperature measurements were carried out on the trapping of interstitial O at dislocation loops in Nb irradiated with 2 × 10^{10} n/cm^2 (1 MeV). The effects of annealing on the 150°C internal friction peak were determined. Annealing above 250°C returns the trapped O to solution. (D.L.C.)

31428 (ORNL-4246, pp 55-66) ANNEALING CHARACTERISTICS OF NEUTRON IRRADIATED NI OBium. Ohr, S. M.; Tucker, R. P.; Bolling, E. D. (Oak Ridge National Lab., Tenn.).

The annealing characteristics were studied using tensile tests (flow yield stress) and electron microscopy (size distribution of defects). Irradiations were carried out to 2 × 10^{19} n/cm^2 (>1 MeV) and 2-hour anneals performed at 200 to 600°C. The observed radiation-anneal hardening is attributed to trapped interstitial C. (D.L.C.)

31429 (ORNL-4246, pp 66-79) FLUENCE DEPENDENCE OF RADIATION HARDENING IN POLYCRYSTALLINE NI OBium. Wechsler, M. S.; Tucker, R. P.; Ohr, S. M. (Oak Ridge National Lab., Tenn.).

Radiation hardening was measured in Nb irradiated with from 7 × 10^{16} to 7 × 10^{19} n/cm^2 (>1 MeV). A plot of the increase in yield stress vs square root of flux shows two straight-line portions, with the less-steep portion occurring at the higher fluxes. Possible explanations for this saturation effect are discussed. (D.L.C.)

31430 (ORNL-4246, pp 75-83) DEFORMATION OF IRRADIATED AND UNIRRADIATED Nb SINGLE CRYSTALS IN COMPRESSION. Guberman, H. D.; Reed, R. E. (Oak Ridge National Lab., Tenn.).

High-purity Nb single crystals were tested (flow stress and ultimate yield strength) at 194, 298, and 529°C before and after irradiation with 1.1 × 10^{14} and 8.3 × 10^{14} n/cm^2 (>1 MeV). The results show a temperature dependence of the yield stress which is somewhat affected by the irradiation. The thermal and mechanical components of the resolved shear stresses are discussed. Activation volume measurements were also made and activation volume vs shear strain curves are given. (D.L.C.)
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From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

(Continued in German).

The magnetization properties and the electrical resistivity of pure superconducting niobium wire samples (resistivity ratio 800) have been studied in a longitudinal magnetic field as a function of reactor irradiation at 4.6 K and subsequent annealing treatment. The irradiation induced a magnetic hysteresis depending on the irradiation dose; the upper critical field $H_c$ increased linearly with the normal resistivity $\rho_H/\rho_0 = 1.56 \, \text{kS/cm}$ in agreement with theory. The pinning force $f_p$ exerted by the defect clusters on the flux lines has been determined by minor magnetization loops. The pinning force $f_p$ as a function of the neutron dose $\phi$ shows a maximum $f_p(\phi_{\text{max}} \sim 10^9 \text{dynes/cm})$ at a dose of $\phi_1 \sim 3 \times 10^{12} \text{n/cm}^2$. The peak effect has been observed in all irradiated and unirradiated specimens exposed in ETR and ALCLAD. The extrapolated saturation value of $\Delta \rho_0 \sim 5 \mu\Omega \cdot \text{cm}$ is in agreement with deuteron irradiation experiments. (auth)


The objective of the Military Compact Reactor (MCR) Irradiation Test Program was to determine experimentally the suitability of candidate fuels to satisfy the requirements of the MCR, and, for cermet fuels, to fulfill the advantages hoped for them. Although the MCR program had scheduled 40 irradiation capsules prior to accepting a fuel material for fabrication into the first core, the results of the initial experiments were so successful, as exhibited by the postirradiation condition of the fuel, that the feasibility of UC$_2$-Nb cermet fuels is felt to have been established. The first experiment showed that ceramic UC$_2$ pellets in a Nb-17 Zr clad is an acceptable fuel, though it lacked the special advantages cited for the cermet. As part of the fuel development program for the MCR, ten high-loaded UC$_2$-Nb cermet and two ceramic UC$_2$ specimens were irradiated in five capsules. Irradiation conditions were representative of MCR design conditions. All capsules performed well, running to their full scheduled terms, and all fuel specimens were in excellent condition after irradiation. Linear specimen growth never exceeded 0.9% and 92 to 99% of all fission gases were retained within the fuel. The microstructure of all cermets was little changed by irradiation although minor cracking did occur. The feasibility of high-loaded UC$_2$-Nb cermets for use in MCR has been established. (auth)

From AEC Corrosion Symposium, Columbus, Ohio.

Results of tests to determine variables affecting irradiation corrosion of Zr alloys are presented. Effects of coolant composition, alloy composition, and pretreatment procedures are emphasized. The data were obtained on samples exposed in ETR and PRTA (J.R.D.)


Physics calculations have shown that the sodium void coefficient of reactivity was 8 percent more favorable for 0.010-inch-thick V-Fe-Zr-C alloy (VANSTAR-7) than for 0.014-inch thick type 316 stainless steel. The Doppler coefficient of reactivity, which was nearly independent of cladding thickness, was 10 percent more favorable for the V-Fe-Zr-C alloy (VANSTAR-7) than for type 316 stainless steel. The breeding ratios at beginning of life and end of life were nearly the same for both vanadium and type 316 stainless steel. Assuming a $4.00/ft tubing cost for vanadium, the fuel cycle cost was about 0.10 mille/kWh higher with vanadium cladding than with type 316 stainless steel at $1.20/ft. Preliminary estimates indicate that, if the core operating temperature is increased by 55°C in the vanadium design and the fuel pellet diameter is increased in both the vanadium and the stainless, the fuel cycle cost decreases by 0.10 mille/kWh. All the vanadium raw material has been received and one ingot of each of the five alloys has been produced. Specifications were developed for VANSTAR-8, VSTAR-9, and the V-20 Ti alloy. Preliminary creep tests indicate that VANSTAR-9 is extremely creep resistant at 700 and 800°C, but that test atmosphere purity may have a very significant effect on creep properties. Studies of the recrystallization and aging behavior of the VANSTAR alloys were initiated. Construction of both sodium corrosion loop systems was completed. The compatibility program was reorganized to include unmodified carbide fuel in the evaluation. The irradiation programs were detailed. (M.C.W.)


Commercial cobalt base alloys including Haynes 25 and UCMA 51 are potential fuel cladding materials at 650°C. These alloys are attractive over type 304 because of their higher strengths. The experimental alloys studied did not show advantages over the commercial alloys with respect to post-irradiation elevated temperature ductility. Cobalt base alloys are susceptible to decreasing ductility with increasing deformation temperature after irradiation. The mode of failure at 800°C is grain boundary tearing similar to that observed in stainless steels and nickel base alloys. (auth)

Effects of neutron irradiation of carbide and boride coatings on high-melting metals were investigated. Carbide coatings cracked and boride coatings disintegrated under radiation doses of 10^19 neutrons/cm^2. The radiation resistance of the coating depended on the radiative expansion of the coating and the base. Tabulated data are given on the radiation resistance of carbide and boride coatings on niobium, molybdenum, tantalum, and tungsten. (R.V.J.)


Research performed by several participating agencies is summarized. Information and data are presented on radiation effects on stainless steel, radiation effects on iron single crystals, effects of hydrogen charging on Hy-80 steel, radiation effects on A302 B steel, annealing EM-IA pressure vessel, radiation effects on Hastelloy alloys, and radiation effects on A 212-B steel. Supporting studies for fast reactors are summarized along with those for ATM gas loop. Investigations of superalloy corrosion are also reported along with tensile tests on SA 533-B steel. (U.R.D.)


The effect of O_2 on "Stage III" annealing in neutron-irradiated Nb was investigated by means of internal friction and electrical resistivity techniques. The O_2 Snook peak decreased during isothermal annealing at 650°C, and the kinetics of this process were similar to that of the electrical resistivity decrease at the same temperature. Oxygen-free samples exhibited no annealing effects in Stage III. It was concluded that Stage III is caused by the migration of O_2 to radiation produced defects. (auth) (UK)

43788 CLADDING AND STRUCTURAL MATERIALS. Reactor Mater., 11: 164-90 (Fall 1968).

The oxidation behavior of Nb-Zr is analyzed. The corrosion behavior of Cr-Fe-Sn-Zr is described. Thixotropic formation and deposition on carbon steels and stainless steels in high-temperature water is discussed. The oxidation behavior of Ta at 500°C and 625°C after annealing in H_2SO_4 is described. Bilateral stress-rupture data for Type 304 stainless steel is presented. The stress-rupture behaviors of Inconel 800 and Cr-Ni-stainless steel at 700°C in contact with Na and air are plotted graphically. Cavitation damage is analyzed for Stellite 6B, L-605, and Type 316 stainless steel in 800°F Na. The effects of liquid Li at 1000°C on the tensile properties of Nb-W, Nb-W-Zr, and Mo-Nb-W-Zr alloys are presented. Radiation effects on the tensile properties of Type 304 and Type 348 stainless steels are analyzed. The effects of radiation on the ductile properties of Type 316 stainless steel in 600°F liquid Na environment are presented. The effect of boron and carbon content on postirradiation elongation of Cr-Ni-stainless steel at 750°C is plotted graphically. The effect of helium on the stress-rupture ductility of Type 304 and modified Type 304L stainless steels irradiated at 50°C is described. The stress-rupture properties of irradiated and unirradiated Inconel 800 at 750°C and Hastelloy X at 650°C are presented. The tensile properties of Hastelloy X-280 and Inconel 600 at 1350°F are listed. Neutron radiations and temperature effects on tangential rupture strength of Cr-Ti-V are described. Radiation effects on the tensile properties of Nb are presented. The effects of stress and atmosphere on the creep rates of Magnes Zr-55 at various temperatures are analyzed. Analysis of hydride orientations for Zircaloy fabrication processes is presented. Tensile properties of Cr-Nb-Zr alloy are described. The effects of fast neutrons on gross volume increase of NbC, TaC, ZrC, TiC, and WC are discussed. The phase diagram for Nb_3O_7-PdO at 700 to 1400°C is presented. 121 references are included. (D.C.C.)


Metals covered are Nb, Ta, W, and Mo. (UK)


Neutron radiation effects and temperature effects on the mechanical properties of Al-Ni-steel-Ti and Mn-Mo-Nb-Ni-steel are analyzed. The steels are possible pressure vessel materials for the Bohunic Power Reactor, Unit 1. Testing procedures and results are described. (D.C.C.)

The irradiation damage configuration and yield stress response obtained in the c.p.h. (a) and (a) phases in a Zr-2.7% Nb alloy after fast-neutron irradiation were studied as a function of pre-irradiation solute distribution. Largest yield stress increments were obtained for specimens irradiated with the solute initially in metastable solution, pre-irradiation ageing treatments reducing the yield stress response. Electron microscope studies showed that the irradiation damage configuration was also determined by the solute super-saturation in the alloy phases, maximum strengthening being associated with a population of defect clusters or loops of \( < 30 \) Å diameter. A mechanism for the defect stabilization is proposed. (auth. UK)


The effect of radiation on the time-temperature-, temperature-, and stress-dependent proportion of selected heat-resisting alloys and refractory metals was determined. Causes of property changes were identified and remedial measures are planned. Materials investigated include A-286 iron alloy; Hastelloy X; Hastelloy H; Hastelloy P; Hastelloy H and Al—Cr—Fe—Y alloys; AISI 304, 316, and 318 stainless steels, ASTM A 302B and A 350-LF3 pressure vessels steels, and various Inconel and Incoloy alloys and heat-resisting materials such as Mo, Nb, Ta, V, and W. (F.S.)


As the effects of radiation on the hardness of Nb and the alloys of Nb with Mo (25 wt %) and Fe (2.0, 3.0, 3.5 and 37 wt %), as well as Nb and various Inconel and Incoloy alloys and heat-resisting materials such as Mo, Nb, Ta, V, and W. (F.S.)

11279 (BNWL-919, pp 12-1-58) IRRADIATION DAMAGE TO REACTOR METALS. Bement, A. L. (Sattelite-Northwest, Richland, Wash. Pacific Northwest Lab.).

Operations in the ETR reactor for Cycle 5 and Cycle 6 are described. The effects of thermomechanical treatments on tensile properties of austenitic stainless steels are presented. The effects of neutron radiation on the mechanical properties of Type 304 stainless steel, Hastelloy X, Zircaloy 2, and Nb—Zr are analyzed. Fracture toughness values for specimens of ASTM A 351 steel are measured. A fast neutron irradiation program is described. Fracture studies for Zircaloy 2 are discussed. The dislocation-defect interactions that provide changes in mechanical properties are described. The effects of stress on irradiation-promoted phase transitions in Zircaloy 2 are discussed. Corrosion and hydrogen absorption of Zircaloy 2 is evaluated for NH 2O environment. Stress-rupture tests for tubing of Nb—Zr are described. Cavity distributions in Type 304 stainless steel are described and influenced by fast neutron irradiation. Fast neutron radiation effects on the tensile properties of Type 304 stainless steel and Type 316 stainless steel are analyzed. A flow diagram for the manufacturing process and quality control inspection of stainless steel tubing is presented. (D.C.C.)


Progress is reported on development of nitride and sial-gel fuels, radiation effects on fission gas release from fuels, Zr metalurgy research, alkali metal corrosion, radiation effects on refractory materials, LMFBR materials research, radiation effects on materials used in nuclear technology, W metallurgy, nondestructive testing, joining nuclear materials, fuel development, SAP development, and metallurgy of refractory metals and alloys. (U.R.D.)
The effects of fast neutron flux, testing temperatures, and fabrication methods on the mechanical properties of several candidate materials for reactor pressure tubes are described and rest results tabulated. These materials include: Zircalloy-2, Zircaloy-4, niobium-tin-Zircaloy-base alloy, niobium-Alxmonum-base alloy, aluminum-Chromium-Iron-base alloy, aluminum-Chromium-Iron-Yttrium-base alloy, and AISI Type 410 stainless steel. The effects of these same variables on the burst strength, creep and stress rupture strength, fatigue crack growth, and crack propagation properties of pressure tubing made from most of these materials are also presented. (L.C.L.)

Transmission electron microscopy was used in a study of the features of dislocation channeling in irradiated and plastically deformed Nb. Electron beam melted monocrystalline Nb was cold rolled into 0.13 mm thick sheet, annealed at 960°C in vacuum, irradiated at a neutron flux of 1.8 x 10^14 neutrons/cm^2 sec^-1 (1 MeV), and deformed under tension at room temperature. The average grain size in the specimens was 44 μ and exhibited a preferred grain orientation with [111] nearly perpendicular to the surface. A measure of the direction and magnitude of shear deformation produced by channeling dislocations was determined by observation of the offset where microstructural features were crossed by dislocation channels. It is postulated that surface slip lines correspond to dislocation channels and that strain is heavily concentrated in the channels. It was observed that there generally appeared to be no defect clusters within cleared channels when the channel plane was perpendicular to the plane of view. The precise mechanisms involved in the removal of defect clusters is a matter of conjecture. A possibility considered is that the motion of channeling dislocations produces an input of energy, largely converted to heat, which may anneal the damage clusters caused by radiation. It is also possible that defect clusters are removed during the motion of channeling dislocation by the production of antidefects. It is considered that the phenomenon of dislocation channeling plays a central role in the plastic deformation of irradiated metals. (P.G.C.)

Neutron-irradiated single crystals of niobium were tensile tested over a range of temperatures from 93 to 298°C. The yield and flow stresses were increased upon irradiation, and the rate of work hardening, the uniform strain, and the fracture strain were reduced. Slip markings on polished surfaces were coarser and more inhomogeneous for the irradiated samples. The increase in the critical shear stress upon irradiation was approximately the same over the entire range of test temperatures, with perhaps a slight tendency for greater hardening at lower temperatures. The shape of the stress-temperature (τ vs. T) curve was analyzed according to several expressions. A reasonably good fit to the Fleischer expression, (f(T)/T) = 1 - (T/T_0)^n, was obtained, but an equally good fit was found to the simpler relation (f/T) = 1 + (T/T_0). The yield stress upon irradiation was increased and the critical shear stress appeared to be a parameter of the temperature dependency of the yield and flow stresses, and the rate of work hardening, the uniform strain, and the fracture strain were reduced. Slip markings on polished surfaces were coarser and more inhomogeneous for the irradiated samples. The increase in the critical shear stress upon irradiation was approximately the same over the entire range of test temperatures, with perhaps a slight tendency for greater hardening at lower temperatures. The shape of the stress-temperature (τ vs. T) curve was analyzed according to several expressions. A reasonably good fit to the Fleischer expression, (f(T)/T) = 1 - (T/T_0)^n, was obtained, but an equally good fit was found to the simpler relation (f/T) = 1 + (T/T_0).
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Neutron irradiation effects on zirconium-base alloys with particular reference to irradiation hardening and its recovery at high temperature were investigated and compared. Zirconium base alloys including Nb-Zr, Cr-Zr, Cu-Nb-Zr, and Cr-Nb-Zr undergo hardening on quenching from β-phase due to a martensitic reaction. In Cr-Nb-Zr, radiation induced hardening in the as-quenched state was lower than in the as-annealed condition. Nimonic alloys including Nb-Zr, Cr-Nb-Zr and tension, Hastelloy X, Incoloy 800, Hastelloy R253, aluminum-iron alloys, A-286 alloy, Hastelloy N, Inconel, niobium base alloys, and iron.

Information is included on reactor pressure vessels, impact shock testing of metals, radiation detection foils, Edit II operation, and in-pile gas loop development. (J.K.M.)

24955 (BNWL-570, pp 11,1-90) RADIATION METALLURGY Billington, D. S.; Wechsler, M. S. (Oak Ridge National Lab., Tenn.).

ANNEALING — effects of post-irradiation, on property changes due to impurities in zirconium metal.

AUTORADIOGRAPHY — method for determining inhomogeneous distribution of titanium and tungsten in zone-refined niobium.

HAUSTELLOY N — radiation effects on creep properties of titanium-modified, thermal neutron.

NIOBUM — radiation effects on microstructure of zone-refined, impurity dependence of, fast-neutron.

— titanium and tungsten distribution in zone-refined, micro-radiographic detection of inhomogeneous.

— property changes in post-irradiation annealed zone-refined, effects of impurities on.

— annealing of irradiated, effects of oxygen on.

— internal friction of hairpins of, effects of measuring temperature on.

— dislocation channels in irradiated and plastically deformed, micrographs of.

OXYGEN — effect on annealing of irradiated niobium.


NIOBUM A & S — Nb-Zr, radiation effects on stress-rupture at 300°C, in-pile.


NIOBUM — magnetic flux pinning at radiodqued defects in superconducting.

— annealing of neutron-irradiated, effects of interstitial impurities on.

— radiation-induced hardening in, studies of defect clusters in fast-neutron.

— impurity distribution in electron-beam-zone-refined, micro-radiography of tantalum and tungsten.

NIOBUM ALLOYS & SYSTEMS — Nb-base-Zr, magnetic flux pinning at radiodqued defects in superconducting.


Results of investigations are presented on neutron effects on stainless steel, steel, cobalt alloys, chromium steel, tungsten, Hastelloy X, Incoloy 800, Hastelloy R253, aluminum-iron alloys, A-286 alloy, Hastelloy N, Inconel, niobium base alloys, and iron.

Information is included on reactor pressure vessels, impact shock testing of metals, radiation detection foils, Edit II operation, and in-pile gas loop development. (J.K.M.)


The development of cold-worked Zr-2 1/2 Nb for use as pressure tube material is described. The mechanical properties of cold-worked Zr-2 1/2 Nb are given and a comparison with cold-worked Zircaloy-2 is made, which shows that the strength of Zr-2 1/2 Nb both at 20 and 300°C is the greater by 20 to 25,000 psi. Irradiation at 30°C to doses of 3.2 x 1012 n/cm2 (Ni) increased the strength of both Zr-2 1/2 Nb and Zircaloy-2 by about 50,000 psi. Creep data for pressure tube material are given for 300°C, which fit the equation $e = 1.45 	imes 10^{-5} a^413$ X-ray and transmission electron microscopy results are presented which show that the observed strengthening in Zr-2 1/2 Nb may be attributed to Nb and O2 in solid solution, to precipitation in the α-Zr phase during cooling, and to cold work. Aging at 400°C and below causes further precipitation, which explains the strengthening observed after autoclaving, while aging above 400°C causes rapid over-aging and softening. (auth) (UK)


The susceptibility of quenched and aged Zr-2.5 wt % Nb alloy to embrittlement during irradiation has been examined for a number of solution temperatures and aging times. Material quenched from temperatures approximately 40°C below the β transition has high tensile ductility, and this ductility is insensitive to aging at 200°C or to irradiation. If, however, the material is quenched from temperatures above the transition it becomes highly susceptible to loss of ductility either from aging at 300°C or from irradiation. Intergranular failure is characteristic of the material having low ductility. The distribution of the equiaxed α phase is found to control the susceptibility to embrittlement by restricting grain growth during heat treatment and thus influencing crack propagation. (auth)


From International Conference on the Strength of Metals and Alloys, Tokyo, Japan. See CONF-670909.

The effect of temperature during neutron irradiation on post-irradiation deformation behavior was studied using an annealed Zr-2.5 wt % Nb alloy. Tensile specimens oriented for deformation by slip were irradiated at 50 to 100°C and 250 to 325°C. After
Irradiation to integrated neutron fluxes of \( \sim 5 \times 10^{16} \) n/cm\(^2\) (E > 1.0 MeV) transmission electron microscopy revealed small clusters or loops \( \sim 25 \) \( \AA \) diameter in material held in the lower temperature range, with some cluster or loops \( \sim 100 \) \( \AA \) diameter in material held in the higher temperature range. The defects generated at \( \sim 300^\circ \)C were responsible both for a larger increment in yield stress and a higher rate of work hardening than the defects formed at the lower temperature. A reasonable correlation is obtained between the changes in deformation characteristics and the defects formed during irradiation. (auth)


Tubes of Zr-2.5 wt \% Nb alloy were fatigue in the tension-shear mode by cyclic internal pressures to cause axial crack growth and unstable fractures at room temperature. Pressure cycle rates were 400 and 3000 cycles per hour. Both the cold worked and heat treated conditions, before and after hydrating \( (100 \pm 10) \) \( \mu \)m, were investigated. Exploratory tests were done to determine the effect of the axial length of the surface stress intensity factor, and multiple surface stress intensifying defects on fatigue crack initiation, growth, and critical length at unstable fracture. From short (less than or equal to tube wall thickness) length defects, fatigue crack initiation and growth will occur at nominal peak hoop stresses equal to or less than the estimated endurance limit (about 25,000 psi). (auth)


Activities in the materials testing program devoted to evaluation of corrosion and mass transfer by liquid Na of potential containment and fuel cladding materials for use at 760 to 815°C are reviewed. The current status of the program is presented tabularly. Research on physical chemistry aspects of Na reactions with Ni and Fe is summarized along with results of investigations of Na impurity chemistry, Na-O chemistry, Na-H reactions, and Na-C reactions. Efforts devoted to development of methods for studying impurities reactions with sodium by electrochemistry are reported along with results of work on Na analytical methods. Work on radiation effects on LMFBR fuel cladding and plastic deformation of bcc metals is summarized along with work on superconductivity, properties of liquid metals, and metallographic techniques. (J.R.D.)


From Conference on Properties and Uses of Highly Pure and Reactive Metals, Dresden, Germany. See CONF-670222. Work done since 1960 on the development of corrosion-resistant Zr-Nb alloys is summarized. The mechanical and corrosion-resistant properties used in the evaluation of the ternary alloy Zr-3 Nb-1 Sn are described. The effects of structure, oxygen content, heat treatment, and neutron irradiation on the properties of binary and ternary Zr-Nb alloys are indicated. Problems connected with the production of some alloys are discussed, and alloys developed for expanded temperature range are considered. (59 references) (J.S.R.)


Studies were made on pre-corroded tubular specimens of zirconium-2 and Zr-2.5% Nb alloy, using the \( ^{13}Xe \) adsorption BET technique at \( 77^\circ \)K. The corroding environments were steam or moist \( CO_2 \) at \( 300^\circ \) to \( 400^\circ \)C. Irradiation was in the DIDO reactor at \( \sim 3 \times 10^{16} \) n/cm\(^2\)/sec, \( \geq 1 \) MeV, and \( 10^5 \) to \( 10^6 \) R/hr by \( \gamma \). Decreases of corrosion were expressed as increase in weight per cm\(^2\). It was found that irradiation consistently increased the surface area. Different behavior was shown by the irradiated specimens as compared with those unirradiated. A relatively short irradiation period increased the surface area dramatically, and the observations were consistent with the suggestion that the corrosion enhancement was associated, at least in part, with irradiation embrittlement of the film, which leads to extensive breakdown and hence increase in internal surface area. (CN)


The dose dependence of the annealing spectra of Ta and Nb from 15 to 285 K after bombardment with 2.3-MeV electrons were investigated. Ta and Nb samples were irradiated to different defect concentrations, then simultaneously annealed (isochronally from 147 up to 285 K). In Ta recovery peaks were observed at 23, 51, 65, 123, and 172 K. The position of the peak near 170 K is dose dependent and shows the characteristics of a second-order reaction. In Nb recovery peaks were observed at 26.3, 58, 80, and 130 K. The peak at 95 K is dose dependent and follows second-order kinetics initially with an apparent shift to higher order toward the end of the stage. A two-interstitial model seems to give the best fit to the data. (auth)
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Changes in the damage rate (residual electrical resistivity per unit electron fluence) were measured as a function of incident electron energy in the range from 1.0 to 2.2 MeV for 0.002-in-diameter Nb and Ta wires. The threshold energies, estimated by simple extrapolation to zero production rate, were 36 eV for Nb and 32 eV for Ta. The resistivity of a Frenkel pair was determined to be about 5.3 μΩ-cm/cm2 for Nb and 1.9 μΩ-cm/cm2 for Ta. The damage production curves for two other bcc metals, Mo and W, were compared to the Nb and Ta production curves. (auth)

42707 (BNL-50150(PL3), pp 987-1132) ACCELERATORS AND STORAGE RINGS USING SUPERCONDUCTING OR CRYOGENIC MAGNETS. (Brookhaven National Lab., Upton, N. Y.);

45591 (BNWL-1144, pp 121-75) IRRADIATION DAMAGE TO REACTOR METALS. Blackburn, L. D. ( Battelle-Northwest, Richland, Wash., Pacific Northwest Lab.).

A total of 162 tensile, creep, and corrosion specimens were discharged from the ETR at the conclusion of Cycle 100. Core heating in the ATR was completed. A comparison was made of the stacking fault hardening rates produced by quenching, straining at room-temperature, straining at 700°C, and straining at 700°C followed by a half-hour hold under load at 700°C. The results showed that hardening-rate increased in the order: straining plus hold > straining > quenching and that straining plus hold treatment produced a faster hardening-rate than did straining alone which was rationalized in terms of the added effect of vacancy-enhanced diffusion. Discontinuous yielding during neutron irradiation was measured in annealed 304 SS at 370°C stressed to 35,000 psi. This phenomenon, induced by irradiation, resulted in an average creep-rate of 6 x 10^-7 hr^-1. The values of the activation parameters and the stress-dependence of the creep-rate in Zr-2.5 Nb are consistent with the "dislocation-climb" model for creep with short-range diffusion. Neutron irradiation can be thought of as adding a stress component for faster creep. The stress arises from an osmotic force on climbing dislocations resulting from the super-saturation of point defects created by irradiation. Irradiation of high purity 304 SS to 2 x 10^11 n/cm^2 (E > 0.1 MeV) at 450°C produces ~1 x 10^15 voids of average diameter ~120 Å. Analysis of small-angle scattering from metals irradiated at elevated temperatures yields void diameters consistent with those measured by electron microscopy. An equation describing the effect of fluence and temperature on swelling in solution-treated 304 SS has been refined. Transmission electron microscopy studies of irradiated SS indicate that the nucleation rate of cavities may increase with accumulated exposure. An analysis of carbide distributions in 316 SS indicates an increase in carbide precipitation in irradiated material. Eight BNWL structural materials experiments were discharged from EBX-11. The flux monitors from these discharged experiments are in the process of being counted. Four BNWL weldment experiment pins in subassembly x-007 are to have a goal exposure of 1 x 10^14 n/cm^2 (E > 0.1 MeV). IRRADIATED TEMPERATURE RISE TESTS AND 1200°F STRESS-RUPTURE TESTS FOR TIMES LESS THAN 1440 HR HAVE BEEN COMPLETED ON TUBING LOCS E, F, G, H. Metallography and check chemical analyses have been completed on tubing loc. G. Biaxial creep curves on tubing locs E & F have been completed. The results of post-irradiation creep-rupture tests on 304 and 316 stainless steel are reported in specimens irradiated in the temperature range of 700 to 900°F and total fluences from 1 x 10^12 n/cm^2 to 1 x 10^13 n/cm^2 (E > 0.1 MeV) by ETR-II. Specimens were strained sufficiently to cause rupture in 0.1 to 0.5 year in unirradiated material. The data suggest some increase in rupture strength for shorter rupture times and loss in rupture strength for longer rupture times for irradiated 304 SS. Strain is observed to be sharply reduced from irradiated values. Characterization of candidate ETR vessel weldment processes is in progress. Tensile test results are reported for an-fabricated specimens from weldments in 1-inch-thick 304 SS plate. The testing matrix includes four welding processes, three specimen types, and two testing temperatures. Data matrix includes the baseline mechanical property information to be used in conjunction with the weldment irradiation experiment. Four MK-B-7 type pins containing tensile, creep-rupture, and fatigue specimens have been shipped to the ETR-II site for inclusion in subassembly X-017. The LMFBR cladding fabrication section of the LMFBR fuel and cladding information center is in operation. (auth)


Trends. An investigation of the dose dependence of the annealing spectra at room temperature of Nb and Ta irradiated to 2 x 10^17 cm^-2 to 2 x 10^19 cm^-2 after bombardment with 2.2 MeV deuterons is reported. Three thin wire Ta samples and two thin wire Nb samples were irradiated at 15°C to different defect concentrations. These samples were then simultaneously annealed isothermally from 117 to 293 K going up in temperature steps of 25°C of the annealing temperature. In Ta recovery peaks were observed at 25, 51, 83, 123, and 170 K. Only two peaks at 25 and 123 K show a definite dose dependence. The annealing of this stage obeys a second order rate equation. Recovery peaks were observed in Nb at 26.3, 58, 85, and 133 K. Higher order recovery is observed for the peak at 93 K. This peak starts out as a second order and then gradually goes to a stilted higher order towards the end of the stage. It is proposed that two mechanisms may seem to give the best fit to the observed data. Kinetic effects appear to be of considerable importance in their suppression of free migration annealing stages. Several stages are also interpreted as interstitial impurity release processes. There is some evidence that the free migration of the 311 split interstitial can be assigned to substages below 200°C in both Ta and Nb, while the 310 split interstitial is thought to be migrating at the more elevated temperatures of 350°C in Nb and 170°C in Ta. (abstract)

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An evaluation is presented of the potential of various classes of alloys for use as the fuel cladding material in a liquid metal cooled, fast breeder reactor. The evaluation is based on reviews of the data on the sodium corrosion resistance, nuclear characteristics, irradiation effects, and creep or creep-rupture properties. The materials considered include the alloys of iron, nickel, molybdenum, tantalum, and tungsten. It was concluded that the alloys of immediate potential are the austenitic steels and the Fe-Cr-Co-X alloys and that those with development potential offering higher creep strengths include the nickel, molybdenum, tantalum, and niobium alloys. The ferritic steels, nickel alloys with less than 50 percent nickel, and the cobalt, vanadium, niobium, molybdenum, tantalum, and tungsten. It was concluded that the alloys of immediate potential are the austenitic steels and the Fe-Cr-Co-X alloys and that those with development potential offering higher creep strengths include the nickel, molybdenum, tantalum, and niobium alloys. The ferritic steels, nickel alloys with less than 50 percent nickel, and the cobalt, vanadium, tantalum and tungsten alloys are classified as of limited interest. Irradiation effects are considered the most important problem area for all of the alloy classes. Other limitations noted included the strength of the austenitic steels, the ductility of the cobalt alloys, corrosion of the nickel, vanadium, and niobium alloys, and the nuclear characteristics of niobium and molybdenum alloys. (auth)

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NEUTRONS, FAST — effects on high-temperature tensile properties of alloys of varying nickel content
- effects on superplastic chromium-iron-nickel alloys
- effects on low carbon, low nitrogen stainless steel
- effects on niobium, Stage III recovery from
- effects on tensile properties of hydrogen-charged steels
- effects on structural steels
- effects on heat-resistant alloys and refractory metals
- effects on iron
- effects on stainless steels in EBR-II
- effects on microstructures of metals

NIOBium — recovery in neutron-irradiated, mechanism of Stage III
NIOBium ALLOYS AND SYSTEMS — Nb- Zr-base, creep rate acceleration factor at 300°C
- Nb- Zr-base, creep-rupture testing of pressure tubing at 300 and 400°C

50980

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The corrosion behavior, mechanical properties, and creep of zirconium alloys are discussed in relation to their use as fuel cladding or pressure tubing in pressurized or boiling water reactors operating at about 300°C. Particular emphasis is placed on the effects of irradiation and hydrogen which is absorbed during service. Most attention is paid to Zircaloy-2 and Zr-2 1/2% Nb but other alloys such as Ozhennite 0.5, Zr-1% Nb, and Zr-3% Nb-1% Sn are briefly mentioned. It is suggested that, although Zircaloy-2 has performed well in actual service, the Zr-2 1/2% Nb alloy offers considerable advantage in terms of higher strength, better in-reactor creep performance, and superior corrosion resistance particularly under boiling water reactor conditions. The use of zirconium alloys at higher temperatures is considered. Ozhennite 0.5 shows high corrosion and hydriding resistance in organic coolants and is a potential fuel cladding and pressure tube material for an organic cooled reactor operating at outlet temperatures of 400 to 425°C. An alloy with better corrosion and hydriding resistance to steam at temperatures of 400 to 500°C than either Zircaloy-2 or Zr-2 1/2% Nb would offer advantages for use as fuel claddings in high steam quality boiling water reactors or possibly even in superheat. Ozhennite 0.5 and Zr-1% Cr-0.1% Fe are presently being considered for this application. (auth)


A differential heat flux calorimeter is described for measuring small heats of reaction and the specific heat of massive samples after fast neutron irradiation at low temperatures (4.6K). The calorimeter can be used at constant or linearly increasing temperature in the range from 4K to 300K. One measures directly the heat flux through the surface of the sample. This type of calorimeter is especially suited to analyze reaction kinetics of stored energy release in crystals after irradiation. The sensitivity ranges from 1.5 x 10^-5 W for T = 370K to 1 x 10^-4 W for T = 4K. For a heat release of 2 mw, the total experimental error is ± 3% at T = 30K and ±18% at T = ± 30K. Measurements of stored energy release in niobium and copper after 4.8K neutron irradiation are given as examples. (auth)