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

Compiled by

Jean Furnish

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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. *(auth)*

688 (AD-646682) **TWINNING AND THE LOW TEMPERATURE MECHANICAL PROPERTIES OF POLYCRYSTALLINE COLUMBIUM, MOLYBDENUM, AND NbMo ALLOYS.** Interim Technical Report, February 1, 1966-January 31, 1967. Shaw, B. J. (Westinghouse Electric Corp., Pittsburgh, Pa.). Contract Nonr-4442(00). 40p. (67-9F7-119-R1). CFSTI.

Results of the mechanical tests, in compression, tension, and by stress relaxation, on niobium neutron-irradiated and non-irradiated molybdenum, and three NbMo alloys are discussed. The tests were made at 200 to -269°C, which covers the ductile-brittle transition temperature of all the polycrystalline materials used. The results indicate that point defects in the crystal lattice inhibit dislocation multiplication mechanisms, thereby increasing the yield and twinning stresses and enhancing the embrittlement. 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)

1070 **NEUTRON IRRADIATION EFFECTS IN SUPERCONDUCTING NIOBIUM.** Kernohan, R. H.; Sekula, S. T. (Oak Ridge National Lab., Tenn.). J. Appl. Phys., 38: 4904-10 (Nov. 1967). (ORNL-P-3181).

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 H_{c2} and small changes in the superconducting transition temperature. In addition, the increase in nonequilibrium behavior of the magnetization with integrated fast-neutron flux establishes that defects are present with the ability to pin magnetic flux. For fast-neutron doses of approximately 2×10^{19} neutrons/cm², the magnetization approaches in behavior that of a completely irreversible type-II superconductor. Successive reductions in cross-sectional area of this 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 sheath approximately 0.1 mm thick. The flux-pinning 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)

4812 (ORNL-4195) **RADIATION METALLURGY SECTION, SOLID STATE DIVISION PROGRESS REPORT FOR PERIOD ENDING JULY 1967.** (Oak Ridge National Lab., Tenn.). Nov. 1967. Contract W-7405-eng-26. 63p. Dep. CFSTI.

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 -50 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 tensile samples indicated an increase in yield stress upon irradiation, followed by a further increase upon annealing in the range -35 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 2×10^{16} neutrons/cm² ($E > 1$ MeV) produced a spot density of approximately 5×10^{15} per cm². The spot size distribution showed a peak at a spot diameter of about 80 Å. 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 1.19 eV. No Stage III 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 $\bar{v} = (\tau/\tau_0)^{m^*}$ was obeyed. For decreasing test temperatures in the range 300°K to 77°K, m^* is not greatly changed and τ_0 is greatly increased. Preliminary results at room temperature indicate a decrease in m^* and an increase in τ_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)

4828 **CREEP MEASUREMENT DURING NEUTRON IRRADIATION.** Gilbert, E. R.; Harding, N. E.; Hood, G. L. (Batelle-Pacific Northwest Lab., Richland, Wash.). Trans. Amer. Nucl. Soc., 10: 503-4 (Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

4831 **DEVELOPMENT AND TESTING OF HIGH-LOADED UO₂-Cb CERMET FUEL.** Fuhrman, N.; Holden, R. B. (United Nuclear Corp., Elmsford, N. Y.). Trans. Amer. Nucl. Soc., 10: 483-4 (Nov. 1967).

From 15th Conference on Remote Systems Technology and Atom Fair, Chicago, Ill., Nov. 5-9, 1967. See CONF-671102.

4839 **DIRECT OBSERVATION OF NEUTRON IRRADIATION DAMAGE IN NIOBIUM.** Tucker, R. P.; Ohr, S. M. (Oak Ridge National Lab., Tenn.). Phil. Mag., 16: 643-6 (Sept. 1967).

Polycrystalline sheets of pure niobium were irradiated to a dose of 2×10^{18} neutrons/cm² ($E > 1$ MeV). Defects appearing as black spots, as well as black-white contrasts, were observed by transmission electron microscopy. The density of the black spots was estimated as 5×10^{15} /cm². The defects were identified as small dislocation loops. Slip traces free of black spots were observed after plastic deformation. Attempts are being made to determine the nature of the loops. (auth)

8801 **RADIATION EFFECTS ON CERTAIN PROPERTIES OF NIOBIUM AND NIOBIUM ALLOYS.** Markovskii, E. A.; Chernyi, V. G.; Tikhonovich, V. I.; Krasnoshechikov, M. M.; Nedyukha, I. M. (Inst. of Metal Physics, Kiev). pp 106-10 of Izucheniye Defektov Kristallicheskogo Stroeniya Metallov i Splavov. Svechnikov, V. N. (ed.). Kiev, Izdatel'stvo Naukova Dumka, 1966. (In Russian).

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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 the material originally contained 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 antifritting characteristics of Nb of electron-beam melts. The change in the character of its interaction with counter-substances appears in friction processes as a result of the increase of the number of defects in its structure. (tr-auth)

11017 ANNEALING SPECTRUM OF ELECTRON-IRRADIATED NIOBIUM AND TANTALUM. El-Salam, Ehab Mohamed Abd. Salt Lake City, Univ. of Utah, 1967. 98p. Thesis.

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-isochronal annealing procedures were used. Tantalum specimens showed six main recovery peaks at temperatures 27.4, 34.2, 40.9, 85.8, 124.5, and 157.0°K with associated activation energies ranging from 0.069 to 0.483 eV and no observed recovery peaks from 157°K to 301°K. Niobium specimens showed seven recovery peaks at temperatures 14.9, 17.8, 24.0, 29.2, 85.8, 99.1, and 135.7°K with associated activation energies ranging from 0.043 to 0.393 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 120 has been irradiated to a fast fluence of 1.37×10^{21} n/cm² ($E > 1$ 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 tests. AISI Types 304 SS, 348 SS, 410 SS, and AM 350 SS were irradiated in the ETR hot water loop at 290°C. Results from tensile tests at 1350°F on Inconel 600 and Hastelloy X 280 specimens held as controls for those irradiated at 1250°F to a fast fluence of 1×10^{20} n/cm² (> 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-rupture tests of Inconel X-750 at 1350°F indicate that irradiation at 540°F to a fast fluence of 1×10^{20} n/cm² decreases rupture life by 68%. 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 unirradiated control test of heat-treated Zr-2.5 wt.% Nb at a stress of 25.6 kg mm⁻² and 300°C. Uniaxial creep tests performed on longitudinal sections of Zircaloy-2 pressure tube material at 31.6 kg mm⁻² 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 A302-B steel. The high pressure forms of Zr, Ce, InTe, CdS, Gd₂O₃, RbCl, and benzene were produced and x-rayed. The high pressure polymorphs of Zr, InTe, and Gd₂O₃ were quenched to ambient condition. The cubic-monoclinic transformation in Gd₂O₃ is reported and the first known x-rays of crystalline benzene 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 209 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 volatile alloy constituents 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 EBR-II. Postirradiation thin film transmission electron microscopy of annealed 304 SS irradiated in the EBR-II has shown that the damage state consists of Frank loops and polyhedral voids. Liquid sodium capsule GEH 22-4 continues operation at peak sodium temperature of 1400°F. The building which will house the testing facilities is in design stage and progress in all areas is reported. (J.C.W.)

13040 METALLURGY AND STRUCTURAL MATERIALS. THEIR ADAPTATION IN NUCLEAR INDUSTRY. Narcy, J. Atompraxis, 13: 527-9(1967). (In German).

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)

15225 (GEMP-1002, pp 38-64) RADIATION EFFECTS ON FAST REACTOR CLADDING AND STRUCTURAL MATERIALS. Moteff, J.; Kingsbury, F.; Smith, J. P. (General Electric Co., Cincinnati, Ohio. Nuclear Materials and Propulsion Operation).

Creep-rupture 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 3×10^{20} n/cm² ($E_g \geq 1$ MeV). Incoloy 800, irradiated to 3×10^{20} n/cm² ($E_g \geq 1$ MeV) and tested at 705°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 1.5 increase in rupture life. Hastelloy R-235 which had been re-solutioned and re-aged following 70°C irradiation exhibited about one-half the as-irradiated rupture life at 870°C. Analysis of creep data for A-286, containing two different B levels and irradiated to various thermal neutron fluences, showed that the reduction in the minimum creep rate is directly related to He formed from the $^{10}\text{B}(n,\alpha)^7\text{Li}$ reaction. The increase in stress dependency, n , ($\dot{\epsilon} \propto \sigma^n$) 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 carbide 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 α particle and Li atom

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recoil distances in Ni. The effect of post-irradiation annealing on the 700°C creep-rupture properties of irradiated Mo (1.1×10^{20} n/cm², $E_n \geq 1$ Mev) has been continued on a series of specimens containing a factor of eight greater C (~200 ppm) content than the previously reported series. Results confirm that an 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° and 1200°C. Initial results of the effects of post-irradiation annealing on the room- and elevated-temperature properties of irradiated Nb sheet (2.1×10^{20} n/cm², $E_n \geq 1$ Mev) show no indication of thermal hardening. A 1-hour anneal at 1000°C was required for complete recovery to unirradiated values, compared with a 600°C anneal reported previously on material irradiated to a lower fluence. Hot-microhardness data were determined for Mo-0.5Ti, Ta, Ta-10W, and 306 alloy (W-30Re-30Mo, at. %), from 25° to 1200°C. A comparison of 256 alloy (W-25Re-30Mo, at. %) hardness and 306 alloy hardness showed the 256 alloy to be harder in the 850° to 1300°C temperature range, a result consistent with the modulus difference between W and Re in the two temperature ranges. A comparison was made of the Mo-0.5Ti data with previous Mo and Mo-TZM data. Radiochemical analysis of ninety-three Fe, Ni, and Ti dosimeters has allowed the preliminary fast-flux mapping of EBR-II rows 2 and 7 positions. Good agreement was obtained between the three types of dosimeters used in the investigation. (auth)

15226 (ORNL-TM-2059) THE PROGRAM OF THE RADIATION METALLURGY SECTION, SOLID STATE DIVISION, OAK RIDGE NATIONAL LABORATORY. Wechsler, M. S. (Oak Ridge National Lab., Tenn.). Feb. 1968. Contract W-7405-eng-26. 49p. Dep. CFSTI.

Radiation damage and radiation hardening and embrittlement in the body-centered cubic metals are discussed. The problem of the radiation embrittlement of 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 bearing on materials problems in nuclear technology are discussed. (auth)

17072 (AECL-2797) EFFECT OF WATER CHEMISTRY ON THE OXIDATION OF ZIRCONIUM ALLOYS UNDER REACTOR RADIATION. Lesurf, J. E.; Bryant, P. E. 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-2½% Nb, oxidized in reactor radiation by water containing excess oxidizing or reducing 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-2½% 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-2½% Nb oxidation other than that associated with oxygen production from the radiolysis of water. Addition of ammonia to the water being irradiated prevents the appearance of molecular oxygen out-reactor and reduces the oxidation of both Zircaloy-2 and Zr-2½% Nb. (auth)

19581 (BNWL-SA-1598) IN-REACTOR CREEP OF ZIRCONIUM ALLOYS. Gilbert, E. R.; Bement, A. L. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Jan. 18, 1968. Contract AT(45-1)-1830. 14p. (CONF-680105-3). Dep. CFSTI.

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 lower creep rates at high stress levels greater than 21 kg/mm² (30,000 psi). At lower stress levels both alloys have approximately the same in-reactor creep rates. (L.C.L.)

19582 (BNWL-SA-1611) IN-REACTOR CORROSION OF ZIRCONIUM ALLOYS. Johnson, A. B. Jr. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Jan. 29, 1968. Contract AT(45-1)-1830. 11p. (CONF-680105-4). Dep. CFSTI.

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 prefilmed. 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.)

19597 REVIEW OF SWEDISH WORK ON IRRADIATION EFFECTS IN CANNING AND CORE SUPPORT MATERIALS. Grounes, M. (Aktiebolaget Atomenergi, Studsvik, Sweden). pp 200-23 of Effects of Radiation on Structural Metals. Philadelphia, American Society for Testing and Materials, 1967.

From 3rd International Symposium on the Effects of Radiation on Structural Metals, Atlantic City. See CONF-660607.

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 7×10^{21} n/cm² (>1 MeV) at 40°C and to 5×10^{19} 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 10^{19} n/cm² (>1 MeV). Tension specimens of annealed and cold-worked steels similar to AISI Types 304 and 316 were irradiated to 4×10^{21} n/cm² (>1 MeV) at 40°C and about 5×10^{20} n/cm² (>1 MeV) at 270°C. Tension specimens of nickel-base alloys (Hastelloy X, Incoloy 800, 20Cr/25Ni/Nb) were irradiated at 550 to 700°C to 10^{20} to 10^{21} n/cm² (>1 MeV). Impact tests were made on AISI 406 steel (13% Cr, 4% Al) after 10^{20} 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)

21729 USE OF ZIRCONIUM ALLOYS IN REACTOR TECHNIQUE. Sedlacek, Vladimir (Metal Research Inst., Prague); Brumovsky, Milan. pp 7-24 of Conference on the Use of Zirconium Alloys in Nuclear Reactors, October 20-21, 1966, Mariánské Lázně, Czechoslovakia. General Proceedings. I. Plzen, Czechoslovakia, Skoda Works, [1966].

See CONF-661059-(Vol.1).

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 Cu or Nb alloy additives are considered. (F.S.)

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21730 INFLUENCE OF NEUTRON IRRADIATION ON THE CORROSION BEHAVIOUR AND THE MECHANICAL PROPERTIES OF Zr-3Nb-1Sn AND ZIRCALOY-2. Richter, H. (Metallgesellschaft A. G., Frankfurt am Main); Ruckdeschel, W.; Spalthoff, W.; Starke, E. pp 111-41 of Conference on the Use of Zirconium Alloys in Nuclear Reactors, October 20-21, 1966, Mariánské Lázně, Czechoslovakia. General Proceedings. I. Plzeň, Czechoslovakia, Skoda Works, [1966].

See CONF-661059-(Vol.1)).

After a review of the corrosion behavior and mechanical properties of ZrNbSn alloys, the in-pile behavior is discussed. Tensile 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.)

21746 RADIATION DAMAGE IN BODY-CENTERED CUBIC METALS AND ALLOYS. Moteff, J. (General Electric Co., Cincinnati, Ohio). Met. Soc. Conf., 37: 727-807(1967).

From Symposium on Radiation Effects, Asheville, N. C. See CONF-650904-(Vol.37).

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)

22830 TESTS OF EXPERIMENTAL ASSEMBLY WITH A ROD TYPE FUEL ELEMENT WITH SINTERED UO_2 CORE IN ZIRCONIUM CLADDING WITH MAXIMUM BURN-OUT OF 68,000 MW DAY/TON URANIUM. Amaev, A. D.; Amvartsumyan, R. S.; Goncharov, V. V.; Glukhov, A. M.; Dubrov, K. P.; Ivanov, E. I.; Lebedev, L. M.; Pravdyuk, N. F.; Ryazantsev, E. P. (Inst. of Atomic Energy, Moscow). pp 143-57 of Conference on the Use of Zirconium Alloys in Nuclear Reactors, October 20-21, 1966, Mariánské Lázně, Czechoslovakia. General Proceedings. I. Plzeň, Czechoslovakia, Skoda Works, [1966]. (In Russian).

See CONF-661059-(Vol.1)).

The results of in-pile and post irradiation tests of an experimental assembly of the icebreaker-type rod fuel element with the sintered UO_2 and Zr-1% 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 1.6×10^{19} 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 1.25×10^{19} 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 1.5 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 Seeger. (auth)

28849 STRESS DEPENDENCE OF DISLOCATION VELOCITY IN SINGLE CRYSTAL NIOBIUM. Guberman, H. D. (Oak Ridge National Lab., Tenn.). Acta Met., 16: 713-21(May 1968).

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 $\bar{v} = (\tau^*/\tau_0)^{m^*}$ where $\bar{v}$ is the average velocity, τ^* is the effective shear stress, and m^* and τ_0 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 8.3×10^{17} 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)

28855 IN-REACTOR CREEP OF Zr-2.5 AT. PERCENT Nb. Gilbert, E. R. (Battelle Memorial Inst., Richland, Wash.). J. Nucl. Mater., 26: 105-11(Apr. 1968).

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 Zircaloy-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 20 kg/mm² 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.)

28858 RADIATION-ENHANCED RELAXATION IN ZIRCALOY-4 AND Zr/2.5 WT PERCENT Nb/0.5 WT PERCENT Cu ALLOYS. Kreyns, P. H.; Burkart, M. W. (Westinghouse Bettis Atomic Power Lab., West Mifflin, Pa.). J. Nucl. Mater., 26: 87-104(Apr. 1968). (WAPD-T-2087).

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 2.4×10^{14} nv > 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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28859 THE IN-REACTOR CREEP OF COLD-WORKED ZIRCALOY-2 AND ZIRCONIUM-2.5 WT PERCENT NIOBIUM PRESSURE TUBES. Ross-Ross, P. A.; Hunt, C. E. L. (Atomic Energy of Canada Ltd., Chalk River, Ont.). J. Nucl. Mater., 26: 2-17 (Apr. 1968).

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^{13} n/cm²·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} /h, 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 $\dot{\epsilon} = \beta \sigma^n$ ($\dot{\epsilon}$ = strain rate, σ = 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 $\dot{\epsilon}_t = 4 \times 10^{-21} \sigma_t \phi (T - 160^\circ\text{C})/\text{hr}$ could be used for cold-worked Zircaloy-2 in the temperature (T) range from 250 to 300°C, hoop stress (σ_t) up to 20,000 psi, and with fast neutron flux (ϕ) up to 3.5×10^{13} n/cm²·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^{13} n/cm²·s fast neutron flux (>1 MeV). (auth)

28860 UNIAXIAL IN-REACTOR CREEP OF ZIRCONIUM ALLOYS. Fidleris, V. (Atomic Energy of Canada Ltd., Chalk River, Ont.). J. Nucl. Mater., 26: 51-76 (Apr. 1968).

Experimental results of 24 uniaxial in-reactor creep tests on various zirconium alloys, covering a range of stresses from 7 to 42 kg/mm² and at 200 to 400°C are shown. At 300°C increases in creep rate of up to an order of magnitude have been measured in a fast neutron flux of $0.4-1.2 \times 10^{13}$ n/cm²·sec ($E > 1$ MeV) with specimens of cold-worked Zircaloy-2. The flux induced increases were smaller with cold-worked and heat treated Zr-2.5 wt % Nb and not measurable in annealed Zr. Prior irradiation had no significant effect on creep of Zircaloy-2 and Zr-2.5 wt % Nb in the 10^{-7} /hr range, but significantly reduced the creep rate in the 10^{-6} /hr and higher ranges. The stress dependence of the in-reactor creep rate appears to follow a power law in cold-worked Zircaloy-2 and Zr-2.5 wt % Nb, with a stress exponent of about 3. The temperature dependence of the in-reactor creep rate changes around 350°C in both cold-worked Zircaloy-2 and heat treated Zr-2.5 wt % Nb. Above 350°C the dependence can be expressed by an apparent activation energy of 50 to 60,000 cal/mol, while below 350°C the activation energy value drops to about 20,000 cal/mol. (auth)

31419 (BNWL-532, pp 11.1-43) 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 1-6) 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.)

31426 (ORNL-4246, pp 42-51) INTERACTION BETWEEN OXYGEN AND RADIATION-PRODUCED DEFECTS IN NIOBIUM. Stanley, J. T.; Williams, J. M.; Brundage, W. E. (Oak Ridge National Lab., Tenn.).

Internal friction-temperature measurements were carried out on the trapping of interstitial O at dislocation loops in Nb irradiated with 2×10^{18} n/cm² (>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 NIOBIUM. Ohr, S. M.; Tucker, R. P.; Bolling, E. D. (Oak Ridge National Lab., Tenn.).

The annealing characteristics were studied using tensile tests (lower yield stress) and electron microscopy (size distribution of defects). Irradiations were carried out to 2×10^{18} n/cm² (>1 MeV) and 2-hour anneals performed at 200 to 800°C. The observed radiation-anneal hardening is attributed to trapped interstitial C. (D.L.C.)

31429 (ORNL-4246, pp 66-75) FLUENCE DEPENDENCE OF RADIATION HARDENING IN POLYCRYSTALLINE NIOBIUM. 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 4×10^{18} n/cm² (>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 525°K before and after irradiation with 1.1×10^{17} and 8.3×10^{17} n/cm² (>1 MeV). The results show a temperature dependence of the yield stress which is somewhat affected by the irradiation. The athermal and thermal 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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31432 CORRELATION OF HOT HARDNESS WITH THE HOT, ULTIMATE TENSILE STRENGTH OF IRRADIATED AND UN-IRRADIATED Nb AND Nb-1Zr. Kamphouse, J. L.; Moteff, J. (General Electric Co., Cincinnati). Trans. Amer. Nucl. Soc., 11: 151(June 1968).

From 14th Annual Meeting of the American Nuclear Society, Toronto. See CONF-680601.

31751 INFLUENCE OF NEUTRON IRRADIATION AT 4.6°K ON THE SUPERCONDUCTING PROPERTIES OF NIOBIUM. Berndt, H.; Kartascheff, N.; Wenzl, H. (Zentralinstitut fuer Tieftemperaturforschung der Bayer, Munich. Technische Hochschule, Munich). Z. Angew. Phys., 24: 305-13(May 1968). (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_{c2} increased linearly with the normal resistivity $\rho_n(\partial H_{c2}/\partial \rho_n = 1.56 \text{ kOe}/\mu\Omega\text{-cm})$ 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 Φt shows a maximum ($f_{p \text{ max}} \sim 10^{-3} \text{ dynes/cm}$) at a dose of $\Phi t \sim 3 \times 10^{18} \text{ n/cm}^2$. The peak effect has been observed in all irradiated the same resistivity change. The mean pinning force on 1 cm flux line per defect cluster is of the order of 10^{-7} dynes . The residual resistivity change $\Delta \rho_n$ shows a tendency towards saturation with irradiation. The extrapolated saturation value of $\Delta \rho_n \sim 3 \mu\Omega\text{-cm}$ is in agreement with deuteron irradiation experiments. (auth)

32924 (GMAD-3078-30) MCR FUEL-IRRADIATION SUMMARY REPORT. Cihl, J.; Kostman, S.; Rush, D. (General Motors Corp., Indianapolis, Ind. Allison Div.). May 3, 1965. Contract AT(30-1)-3078. 156p. (EDR-4215). Dep. CFSTI.

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 loading, the results of the initial experiments were so successful, as exhibited by the postirradiation condition of the fuel, that the feasibility of UO_2 -Nb cermet fuels is felt to have been established. The fifth experiment showed that ceramic UO_2 pellets in a Nb-1% 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 UO_2 -Nb cermet and two ceramic UO_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 UO_2 -Nb cermets for use in MCR has been established. (auth)

34009 (BNWL-SA-1719) PARAMETERS WHICH INFLUENCE THE IN-REACTOR CORROSION OF ZIRCONIUM ALLOYS. Johnson, A. B. Jr. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). May 2, 1968. Contract AT(45-1)-1830. 14p. (CONF-680510-1). Dep. CFSTI.

From AEC Corrosion Symposium, Columbus, Ohio.

Results of tests to determine variables affecting radioinduced 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 PRTR. (J.R.D.)

34046 (WARD-3791-17) VANADIUM CLADDING ALLOY DEVELOPMENT. Quarterly Progress Report for the Period Ending March 31, 1968. Bishop, E. C.; Nadler, R. A.; Whitlow, G. A. (Westinghouse Electric Corp., Madison, Pa. Advanced Reactors Div.). May 1968. Contract AT(30-1)-3791. 66p. Dep. CFSTI.

Physics calculations have shown that the sodium void coefficient of reactivity was 8 percent more favorable for 0.010-inch-thick V-Cr-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-Cr-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 alloys and type 316 stainless steel. Assuming a \$4.00/ft tubing cost for vanadium, the fuel cycle cost was about 0.10 mills/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 reference designs, plant equipment and fuel cycle costs for the vanadium design will be at a break-even point with type 316 stainless steel. All the vanadium raw material has been received and one ingot of each of the five alloys has been produced. Specimens were produced from VANSTAR-8, VANSTAR-9, and the V-20 Ti alloys. 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 replanned to include unmodified carbide fuel in the evaluation. The irradiation programs were detailed. (J.C.W.)

34077 (BNL-12438) EFFECT OF IRRADIATION ON MECHANICAL PROPERTIES OF COBALT BASE ALLOYS. Chow, J. G. Y. (Brookhaven National Lab., Upton, N. Y.). [1968]. 10p. (CONF-680604-3). Dep. CFSTI.

From Symposium on Radiation Effects to Reactor Metals. San Francisco, Calif.

Commercial cobalt base alloys including Haynes 25 and UMCO 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)

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34127 NEUTRON IRRADIATION EFFECTS ON DIFFUSION CARBIDE AND BORIDE COATINGS ON HIGH-MELTING METALS. Ogorodnikov, V. V.; Koval'chenko, M. S.; Krainin, A. G.; Epik, A. P. (Inst. of Problems in Material Studies, Kiev). *Fiz.-Khim. Mekh. Mater.*, 4: 36-8(1968). (In Russian).

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^{20} neutr/cm². 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.)

43615 (BNWL-710) IRRADIATION EFFECTS ON REACTOR STRUCTURAL MATERIALS. Quarterly Progress Report, November-December 1967 and January 1968. Claudson, T. T. (ed.). (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Apr. 1968. Contract AT(45-1)-1830. 257p. Dep. CFSTI.

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 SM-1A 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 ATR gas loop. Investigations of superalloy corrosion are also reported along with tensile tests on SA 533-B steel. (J.R.D.)

43625 EFFECT OF OXYGEN ON "STAGE III" ANNEALING IN NEUTRON-IRRADIATED NIOBIUM. Williams, J. M.; Brundage, W. E.; Stanley, J. T. (Oak Ridge National Lab., Tenn.). *Metal Sci. J.*, 2: 100-4(May 1968). (ORNL-P-3394).

The effect of O₂ on "Stage III" annealing in neutron-irradiated Nb was investigated by means of internal friction and electrical resistivity techniques. The O₂ Snoek peak decreased during isothermal annealing at 150°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 O₂ to radiation produced defects. (auth) (UK)

45798 RADIATION RESISTANCE OF ZIRCONIUM ALLOYS WITH 1 PERCENT NIOBIUM IN CM-2 REACTOR. Votinov, S. N.; Guseva, T. M.; Klimenkov, V. I. *At. Energy. (USSR)*, 24: 473-4(May 1968). (In Russian).

Mechanical properties of zirconium alloys with 1% Nb irradiated with up to 6.9×10^{19} n/cm² and up to 1.1×10^{20} n/cm² at 20°C are tabulated. The shock viscosity of unirradiated specimens and specimens irradiated by fast neutrons up to 5.1×10^{20} n/cm² was 6.7 and 6.2 kgm/cm², respectively. Data obtained on the performance of reactor parts in complex mechanical tension, neutron irradiation, and effects of media proved to be in agreement with published data and confirmed the high resistance of Zr-1% Nb alloy. (R.V.J.)

49788 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. The oxide formation and deposition on carbon steels and stainless steels in high-temperature water is discussed. The oxidation behavior of Ta at 500 to 625°C after anodizing in H₃PO₄ is described. Biaxial stress-rupture data for Type 304 stainless steel is presented. The stress-rupture behaviors of Incoloy 800 and Cr-Ni-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°C 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 Incoloy 800 at 705°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 Magnox 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 Nd₂O₃-PdO at 700 to 1400°C is presented. 121 references are included. (D.C.C.)

49681 BIBLIOGRAPHY OF REFRACTORY METALS. Syre, R. (comp.). AGARD Bibliography No. 5. Maidenhead, Eng., Technivision Services, 1968. 448p. 135s.
Metals covered are Nb, Ta, W, and Mo. (UK)

50748 (ZJE-27) RADIATION DAMAGE AND SURVEILLANCE PROGRAMS FOR CZECHOSLOVAK REACTOR STEEL PRESSURE VESSELS. Brumovsky, Milan (Skoda Works, Pilsen (Czechoslovakia). Nuclear Power Plant Div.). July 1968. 24p. Dep.

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 Bohunice Power Reactor, Unit 1. Testing procedures and results are described. (D.C.C.)

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1013 THE INFLUENCE OF NIOBIUM IN IRRADIATION STRENGTHENING OF DILUTE Zr-Nb ALLOYS. Williams, C. D.; Ellis, C. E. (Chalk River Nuclear Labs., Ontario). *Phil. Mag.*, 18: 763-72 (Oct. 1968).

The irradiation damage configuration and yield stress response obtained in the c.p.h. α' and α 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 $<30 \text{ \AA}$ diameter. A mechanism for the defect stabilization is proposed. (auth) (UK)

6705 (GEMP-1004, pp 89-177) RADIATION EFFECTS ON FAST REACTOR CLADDING AND STRUCTURAL MATERIALS. Motoff, J.; Kingsbury, F. D.; Smith, J. P. (General Electric Co., Cincinnati, Ohio). *Nuclear Materials and Propulsion Operation*.

The effect of radiation on the time-, temperature-, and stress-dependent properties of selected heat-resistant 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 N; Hastelloy R-235; Al-Cr-Fe-Y alloys; AISI 304, 316, and 349 stainless steels, ASTM-A 302B and A 350-LF3 pressure vessel steels, and various Inconel and Incoloy alloys and heat-resisting metals such as Mo, Nb, Ta, V, and W. (F.S.)

8256 EFFECT OF RADIATION ON CERTAIN CHARACTERISTICS OF NIOBIUM AND ITS ALLOYS. Markovskii, E. A.; Chernyi, V. G.; Tikhonovich, V. I.; Krasnoshchekov, M. M.; Ned-yukha, I. M. (Inst. of Metal Physics, Kiev). pp 106-10 of *Izucheniye Defektov Kristallicheskogo Stroyeniya Metallov i Splavov*. Kiev, Izdatel'stvo Naukova Dumka, 1966. (In Russian).

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 on the antifriction properties of Nb, were studied. The specimens were irradiated in an atomic reactor with an integral neutron flux of 10^{19} neutrons/cm² or with γ -quanta from a γ -gun in a total dose of 5.3×10^5 R. Vickers hardness was measured by means of a diamond pyramid under a load of 20 kg and wear tests were performed in a laboratory setup. It was found that irradiation with γ -quanta does not affect the hardness of Nb and Nb-Mo alloy. Irradiation with neutrons on the other hand, affects the mechanical properties (hardness) of Nb and its alloys depending on their purity or chemical composition. Thus, the hardness of pure Nb (obtained from an electron-beam furnace) then nearly doubled, whereas the hardness of Nb containing 0.5% impurities remained nearly unchanged. The hardness of the Nb-Fe alloys containing 2.0, 3.0, and 3.5 wt % Fe increased slightly. On the other hand, irradiation of the Nb-Fe alloy containing 37 wt % Fe produced no change in its hardness. These differences in the hardness of the investigated materials following irradiation apparently are associated with the difference in the degree of their structural defects prior to irradiation: the least number of defects exists in pure Nb obtained from an electron-beam furnace;

in case irradiation produces a large number of defects, which results in the increase in hardness. Apparently the neutron flux having the energy specified above produces additional defects in the material; if the material originally already contained a more or less large number of defects, as in the case of impure Nb and Nb-Mo alloys, neutron irradiation cannot produce many additional defects. As for antifriction properties, neutron irradiation somewhat changed these properties with specimens; this apparently is also attributable to the formation of additional structural defects in the specimens and the attendant change in the pattern of interaction between the surfaces of Nb and the body with which it is in friction. (ATD)

10241 1968 PORTEVIN LECTURE SPECIAL STEELS AND NUCLEAR RADIATIONS. Castro, R. *Mem. Sci. Rev. Met.*, 65: 533-42 (July-Aug. 1968). (In French).

Permanent changes due to neutron radiation, changes due to atom displacement (structural perturbation, saturation effect, order-disorder transformations, phase shifts, accelerated precipitation, resistivity, density, mechanical properties) and changes due to decay are summarized. Radiation effects on the mechanical properties of C-Cr-Nb-Ni-steel are given. (J.C.W.)

11279 (BNWL-919, pp 12.1-58) IRRADIATION DAMAGE TO REACTOR METALS. Bement, A. L. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

Operations in the ETR reactor for Cycle g5 and Cycle g6 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 A533 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 of materials in neutron irradiation are analyzed. The effects of stress on irradiation-promoted phase transitions in ZrO_2 are discussed. Corrosion and hydrogen absorption of Zircaloy 2 is evaluated for NH_4OH environment. Stress-rupture tests for tubing of Nb-Zr are described. Cavity distributions in Type 304L stainless steel are described as influenced by fast neutron radiation. 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. (36 references are included) (D.C.C.)

12260 (ORNL-4330) FUELS AND MATERIALS DEVELOPMENT PROGRAM. Quarterly Progress Report for Period Ending June 30, 1968. Patriarca, P.; Harms, W. O. (comps.); Rucker, D. J. (ed.) (Oak Ridge National Lab., Tenn.). Nov. 1968. Contract W-7405-eng-26. 366p. Dep. CFSTI.

Progress is reported on development of nitride and sol-gel fuels, radiation effects on fission gas release from fuels, Zr metallurgy 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. (J.R.D.)

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16424 (BNWL-746, pp 12.1-38) OTHER PRESSURE TUBE MATERIALS. Pankaskie, P. J.; Gilbert, E. R.; Wheeler, K. R.; Shoher, F. R. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

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 test results tabulated. These materials include: Zircaloy-2, Zircaloy-4, niobium-tin-Zirconium-base alloy, niobium-zirconium-base alloy, aluminum-chromium-iron-base alloys, 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.)

18250 (ORNL-TM-2480) TRANSMISSION ELECTRON MICROSCOPY AND PLASTIC DEFORMATION IN NEUTRON-IRRADIATED NIOBIUM. Tucker, R. P. (Oak Ridge National Lab., Tenn.). Mar. 1969. Contract W-7405-eng-26. 31lp. Dep. CFSTI.

Thesis. Submitted to Univ. of Tennessee, Knoxville.

Defect clusters and plastic deformation in neutron-irradiated polycrystalline and single-crystal niobium were studied by transmission electron microscopy (TEM) and tensile testing to determine the nature of radiation-produced defect clusters (spot damage), the crystal geometry and mechanism of dislocation channeling, the dose dependence of the yield stress, the influence of low-temperature post-irradiation annealing on the yield stress, and the temperature dependence of yielding. Niobium samples were neutron-irradiated at $\sim 50^\circ\text{C}$ for polycrystals, $90-110^\circ\text{C}$ for single crystals to doses, Φ , from 4×10^{15} to 8×10^{18} neutrons/cm² ($E > 1$ MeV). Defect clusters were observed by transmission electron microscopy in polycrystalline niobium over a range of doses from 2×10^{17} to 4.4×10^{18} neutrons/cm² ($E > 1$ MeV). Dislocation channels cleared of radiation-produced defect clusters were observed by TEM in polycrystalline niobium irradiated to doses greater than about 10^{18} neutrons/cm² and then deformed in tension. Mechanisms for the clearing of the defect clusters are described. The yield stress in single-crystal and polycrystalline niobium was found to be proportional to the square root of the dose for $\Phi \lesssim 10^{18}$ neutrons/cm²; at higher doses, the rate of hardening decreased markedly, i.e., a saturation effect was observed. Low-temperature post-irradiation annealing at 175°C and 300 to 400°C gave rise to further yield stress increases (radiation-anneal hardening). The temperature dependence of the yield stress is not substantially changed by irradiation to 2.8×10^{17} neutrons/cm² for test temperatures from 93 to 298°K , which suggested that irradiation to this dose does not change the thermally-activated mechanism of deformation in single-crystal niobium. It is concluded that radiation hardening in niobium can be explained on the basis of the short-range elastic interaction between radiation-produced defect clusters visible by TEM and slip dislocation. (auth)

20519 (ORNL-4334, pp 9-20) EFFECT OF INTERSTITIAL IMPURITIES ON POSTIRRADIATION ANNEALING PHENOMENA IN NIOBIUM. Williams, J. M.; Stanley, J. T.; Brundage, W. E. (Oak Ridge National Lab., Tenn.).

Results of annealing at 50 to 550°C are reported. Impurity levels, analyzed by chemical and physical methods, are reported along with measurements of internal friction and electric conductivity. Discussions are included concerning the reasons for electric conductivity and internal friction variations. (J.R.D.)

20520 (ORNL-4334, pp 21-9) PSEUDO SNOEK PEAK IN NIOBIUM. Stanley, J. T. (Oak Ridge National Lab., Tenn.).

An investigation of the anomalous behavior of the oxygen Snoek peak which developed in irradiated and annealed Nb during internal friction and electric resistivity measurements is described. Evidence is presented to show the Snoek peak pseudo. (P.G.C.)

20522 (ORNL-4334, pp 33-63) DISLOCATION CHANNELING IN NEUTRON-IRRADIATED NIOBIUM. Tucker, R. P.; Wechsler, M. S. (Oak Ridge National Lab., Tenn.).

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 1050°C in vacuum, irradiated at a neutron flux of 1.8 and 4.4×10^{13} neutrons cm⁻² sec⁻¹ (> 1 MeV), and deformed under tension at room temperature. The average grain size in the specimens was 44μ and exhibited a preferred grain orientation with $[111]$ nearly perpendicular to the surface. A measure of the direction and magnitude of shear displacement 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 dislocation 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.)

22595 (BNWL-790, pp 12.1-66) IRRADIATION DAMAGE TO REACTOR METALS. Bement, A. L. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

NIOBIUM A & S—Nb—Zr, radiation effects on creep of, at 315°C , fast neutron

22603 RADIATION HARDENING IN SINGLE CRYSTAL NIOBIUM: THE TEMPERATURE DEPENDENCE OF YIELDING. Wechsler, M. S. (Oak Ridge National Lab., Tenn. Univ. of Tennessee, Knoxville); Tucker, R. P.; Bode, R. Acta Met., 17: 541-51(May 1969).

Neutron-irradiated single crystals of niobium were tensile tested over a range of temperatures from 93 to 298°K . 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, $(\tau/\tau_0)^{1/2} = 1 - (T/T_0)^{1/2}$, was obtained, but an equally good fit was found to the simpler relation $(\tau/\tau_0) = 1 + (T_0/T)$. It is suggested that radiation-produced obstacles exert long-range stresses, giving rise to an athermal inhibition to dislocation motion. A simple analytical treatment is presented that illustrates conditions under which the radiation-hardening is expected to be athermal. (auth)

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24941 (BARC-360) RADIATION HARDENING EFFECTS IN Zr-BASE ALLOYS. Ray, S.; Sharma, B. D. (Bhabha Atomic Research Centre, Bombay (India)). 1968. 12p. Dep.

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. Niobium addition to zirconium base alloys stabilizes radiation damage. Percentage change in room temperature tensile properties of Zircaloy-2, Cu-Nb-Zr, and Cr-Nb-Zr and tensile property changes in irradiated quenched or/and annealed Cu-Nb-Zr, Nb-Zr, and Cr-Nb-Zr alloys and the effect of irradiation on the tensile properties of annealed Cr-Nb-Zr alloys are given. (F.S.)

24955 (BNWL-870, 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 zone-refined niobium
- AUTORADIOGRAPHY—method for determining inhomogeneous distribution of titanium and tungsten in zone-refined niobium
- HASTELLOY N—radiation effects on creep properties of titanium—modified, thermal neutron
- NIOBium—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

24956 (BNWL-870, pp 12.1-55) IRRADIATION DAMAGE TO REACTOR METALS. Bement, A. L. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.).

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

30612 (ORNL-4408) SOLID STATE DIVISION ANNUAL PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1968. (Oak Ridge National Lab., Tenn.). Apr. 1969. Contract W-7405-eng-26. 117p. Dep. CFSTI.

- NIOBium—magnetic flux pinning at radioinduced defects in superconducting
 - annealing of neutron-irradiated, effects of interstitial impurities on
 - radiation-anneal hardening in, studies of defect clusters in fast-neutron
 - impurity distribution in electron-beam-zone-refined, autoradiography of tantalum and tungsten
- NIOBium ALLOYS & SYSTEMS—Nb-base-Zr, magnetic flux pinning at radioinduced defects in superconducting

32040 (BNWL-676) IRRADIATION EFFECTS ON REACTOR STRUCTURAL MATERIALS. Quarterly Progress Report, August-October 1967. Claudson, T. T. (ed.) (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Feb. 1967. Contract AT(45-1)-1630. 336p. Dep. CFSTI.

Results of investigations are presented on radiation effects on stainless steel, steel, cobalt alloys, chromium steel, tungsten, Hastelloy X, Incoloy 800, Hastelloy R235, 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, EBIT II operation, and in-pile gas loop development. (J.R.D.)

32042 (TRG-Report-1786) THE STRENGTH OF Zr-2½% Nb ALLOY IN THE ANNEALED AND COLD-WORKED CONDITION. Murgatroyd, R. A.; Winton, J. (United Kingdom Atomic Energy Authority, Culcheth (England). Reactor Group). Oct. 22, 1968. 19p. (CONF-670909-10; HWR-FEMWG/P-(68)355). Dep.

From International Conference on the Strength of Metals and Alloys, Tokyo, Japan.

The development of cold-worked Zr-2½% Nb for use as pressure tube material is described. The mechanical properties of cold-worked Zr-2½% Nb are given and a comparison with cold-worked Zircaloy-2 is made, which shows that the strength of Zr-2½% Nb both at 20 and 300°C is the greater by 10 to 20,000 psi. Irradiation at 300°C to doses of 5.2×10^{20} n/cm² (Ni) increased the strength of both Zr-2½% Nb and Zircaloy-2 by about 30,000 psi. Creep data for pressure tube material are given for 300°C, which fit the equation $\epsilon = 1.45 \times 10^{-6} s^{2.5} t^{1/2}$. X-ray and transmission electron microscopy results are presented which show that the observed strengthening in Zr-2½% 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)

34035 BETA EMBRITTLEMENT OF THE Zr-2.5 WT PERCENT Nb(Co) ALLOY. Ellis, C. E.; Williams, C. D. (Chalk River Nuclear Labs., Ont.). Trans. Met. Soc. AIME, 245: 1321-8 (June 1969).

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 transus has high tensile ductility, and this ductility is insensitive to aging at 500°C or to irradiation. If, however, the material is quenched from temperatures above the transus it becomes highly susceptible to loss of ductility either from aging at 500°C or from irradiation. Intergranular failure is characteristic of the materials having low ductility. The distribution of the equilibrium α phase is found to control the susceptibility to embrittlement by restricting β grain growth during heat treatment and thus influencing crack propagation. (auth)

34051 EFFECT OF TEMPERATURE DURING NEUTRON IRRADIATION ON SUBSEQUENT DEFORMATION IN THE Zr-2.5 WT PERCENT Nb ALLOY. Ellis, C. E.; Williams, C. D. (Atomic Energy of Canada Ltd., Chalk River, Ont.). Trans. Jap. Inst. Metals, 9: Suppl., 214-19 (1968).

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

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irradiation to integrated neutron fluxes of $\sim 5 \times 10^{19}$ n/cm² ($E > 1.0$ MeV) transmission electron microscopy revealed small clusters or loops ~ 25 Å diameter in material held in the lower temperature range, with some cluster or loops ~ 100 Å diameter in material held in the higher temperature range. The defects generated at $\sim 300^\circ\text{C}$ 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)

35854 (BNWL-1076) FATIGUE CRACK GROWTH AND PROPAGATION IN Zr-2.5 WT PERCENT Nb ALLOY PRESSURE TUBING. Pankaskie, P. J. (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). June 1969. Contract AT(45-1)-1830. 20p. Dep. CFSTI.

Tubes of Zr-2.5 wt % Nb alloy were fatigued in the tension-tension 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 hydriding (200 to 300 ppm H₂), were investigated. Exploratory tests were done to determine the effect of the axial length of the surface stress intensifying defect, 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). For fatigue cracks initiated at the outer diameter surface, the shape of the crack front is semicircular. The number of stress cycles required to initiate fatigue cracking at the surface is inversely proportional to the square of the surface defect length. Crack growth rates for nonhydrided tubing are proportional to ΔK to the fifth power and to the fourth power after hydriding. There was a tendency for the critical crack length at unstable fracture to decrease with an increase in the number of fatigue cycles required to initiate fatigue cracking. Multiple defects may act in concert to initiate unstable fracture. The fracture toughness of the heat treated tubing before hydriding is less than for the cold worked condition either before or after hydriding; after hydriding the fracture toughness is further reduced. In the 30% cold worked condition, irradiation to 10^{20} nvt has little effect unless hydrided; when hydrided, fracture toughness is reduced substantially. (auth)

36155 STRESS DEPENDENCE OF THE IN-REACTOR UNI-AXIAL CREEP OF Zr-2 AND Zr-2.5 Nb. Gilbert, E. R.; Mastel, B. (Battelle-Northwest, Richland, Wash.). Trans. Amer. Nucl. Soc., 12: 132-3 (June 1969).

From 15th Annual Meeting of the American Nuclear Society, Seattle, Wash. See CONF-690609.

41528 (BNL-50149, pp 89-122) METALLURGY AND MATERIALS SCIENCE DIVISION. Gurinsky, D. H. (Brookhaven National Lab., Upton, N. Y.).

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 re-

viewed. 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.)

41635 DEVELOPMENT OF CORROSION-RESISTANT ZIRCONIUM-NIOBIUM ALLOYS WITH HIGH STRENGTH FOR WATER-COOLED NUCLEAR REACTORS. Ruckdeschel, W.; Wincierz, P. (Metallgesellschaft AG., Frankfurt am Main). pp 126-58 of Eigenschaften und Anwendung Hochschmelzender und Reaktiver Metalle. Leipzig. Veb Deutscher Verlag fuer Grundstoffindustrie, 1968. (In German).

From Conference on Properties and Uses of Highly Fusible 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 semifinished alloys are discussed, and alloys developed for expanded temperature range are considered. (59 references) (J.S.R.)

41685 EFFECT OF REACTOR IRRADIATION ON THE INTERNAL SURFACE AREA OF CORROSION FILMS ON ZIRCONIUM ALLOYS. Asher, R. C.; White, J. F. (Atomic Energy Research Establishment, Harwell, Eng.). Nature (London), 223: 497 (Aug. 2, 1969).

Studies were made on pre-corroded tubular specimens of Zircaloy-2 and Zr/2.5% Nb alloy, using the ¹³³Xe adsorption BET technique at 77°K. The corroding environments were steam or moist CO₂/air at 300° to 480°C. Irradiation was in the DIDO reactor at $\sim 3 \times 10^{13}$ n/cm²/sec, > 1 MeV, and 10^6 to 10^8 R/hr γ. Degrees of corrosion were expressed as increase in weight per dm². 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. (UK)

42391 (COO-1494-6) CONCENTRATION DEPENDENCE OF FRENKEL PAIR DEFECT ANNEALING IN ELECTRON IRRADIATED Ta AND Nb. Myhra, Sverre; DeFord, John W. (Utah Univ., Salt Lake City. Dept. of Physics). [1968]. Contract AT(11-1)-1494. 34p. Dep. CFSTI.

The dose dependence of the annealing spectra of Ta and Nb from 15 to 285°K after bombardment with 2.2-MeV electrons were investigated. Ta and Nb samples were irradiated to different defect concentrations, then simultaneously annealed isochronally from 14.7 up to 285°K. In Ta recovery peaks were observed at 23, 51, 65, 123, and 170°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 26.5, 58, 95, and 135°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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42392 (COO-1494-7) THRESHOLD DISPLACEMENT ENERGY MEASUREMENTS ON Ta AND Nb. Youngblood, G.; Myhra, S.; DeFord, J. W. (Utah Univ., Salt Lake City, Dept. of Physics). [1968]. Contract AT(11-1)-1494. 30p. Dep. CFSTI.

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 $5.3 \mu\Omega\text{-cm/at.}\%$ for Nb and $1.9 \mu\Omega\text{-cm/at.}\%$ for Ta. The damage production curves for two other bcc metals, Mo and W, were compared to the Nb and Ta production curves. (auth)

42737 (BNL-50155(Pt.3), pp 967-1132) ACCELERATORS AND STORAGE RINGS USING SUPERCONDUCTING OR CRYOGENIC MAGNETS. (Brookhaven National Lab., Upton, N. Y.).

NIORIUM ALLOYS AND SYSTEMS—niobium-tin radiation effects on intermetallic compound Nb_3Sn , used in superconducting magnets, deuteron, neutron, and proton
—niobium-zirconium, radiation effects on intermetallic compound Nb_3Zr used in superconducting magnets, deuteron and neutron
—niobium-titanium, radiation effects on, used in superconducting magnets, deuteron

45091 (BNWL-1144, pp 12.1-75) 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 I testing 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 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 $6 \times 10^{-5} \text{ hr}^{-1}$. 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-circuited 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 supersaturation of point defects created by irradiation. Irradiation of high purity Fe to $3 \times 10^{21} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$) at 450°C produces $\sim 1 \times 10^{14}$ voids of average diameter $\sim 280 \text{ \AA}$. 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 indicates that the nucleation rate of cavities may increase with accumulated exposure. An analysis of carbide distributions in 316 SS indicates an increase of carbides precipitation in irradiated material. Eight BNWL structural materials experiments were discharged from EBR-II. The flux monitors from these discharged

experiments are in the process of being counted. Four BNWL weldment experiment pins in subassembly X-067 are to have a goal exposure of $1 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$). Elevated temperature burst tests and 1200°C stress-rupture tests for times less than 1400 hr have been completed on tubing lots E, F, G, & H. Metallography and check chemical analyses have been completed on tubing lot G. Biaxial creep curves on tubing lots E & F have been completed. The results of postirradiation creep-rupture tests on 304 and 316 stainless steel are reported for specimens irradiated in the temperature range of 700 to 900°F and to total fluences from $1 \times 10^{21} \text{ n/cm}^2$ to $1 \times 10^{22} \text{ n/cm}^2$ (EBR-II calculated). Rupture-life reductions were observed for all irradiated specimens. Both increased and decreased creep-rates were observed depending on temperature and stress-level of test and irradiation fluence. Biaxial stress-rupture tests at 1200°F have been initiated on AISI type 304 SS irradiated to $0.8 \times 10^{22} \text{ n/cm}^2$ ($E > 0.1 \text{ MeV}$) in EBR-II. Specimens were stressed sufficiently to cause rupture in 0.1 to 1000 hr in unirradiated material. The data suggest some increase in rupture strength for short rupture times and loss in rupture strength for long rupture times for irradiated 304 SS. Strain is observed to be sharply reduced from unirradiated values. Characterization of candidate FTR vessel weldment processes is in progress. Tensile test results are reported for as-fabricated specimens from weldments in 1-inch-thick 304 SS plate. The testing matrix includes four welding processes, three specimen types, and four testing temperatures. This matrix constitutes the baseline mechanical property information to be used on conjunction with the weldment irradiation experiment. Four MK-B-7 type pins containing tensile, creep-rupture, and fatigue specimens have been shipped to the EBR-II site for inclusion in subassembly X067-4F2. The LMFBF cladding fabrication section of the LMFBF fuel and cladding information center is in operation. (auth)

45096 CONCENTRATION DEPENDENCE OF FRENKEL PAIR DEFECT ANNEALING IN ELECTRON-IRRADIATED Ta AND Nb. Myhra, Sverre. Salt Lake City, Univ. of Utah, 1969. 113p.

Thesis.

An investigation of the dose dependence of the annealing spectra of Ta and Nb from 15 to 285°K after bombardment with 2.2 MeV electrons is reported. Three thin wire Ta samples and two thin wire Nb samples were irradiated at 14°K to different defect concentrations. These samples were then simultaneously annealed sequentially from 14.7 to 285°K going up in temperature steps of usually 5° of the annealing temperature. In Ta recovery peaks were observed at 23, 51, 65, 123, and 170°K . Only the peak at 170°K shows definite dose dependence. The annealing of this substage obeys a second order rate equation. Recovery peaks were observed in Nb at 26.5, 58, 95, and 135°K . Higher order recovery is observed for the peak at 95°K . This peak starts out obeying second order and then gradually goes to still higher orders towards the end of the stage. It is proposed that a two-interstitial model seems to give the best fit to the observed data. Impurity 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 $\langle 111 \rangle$ split interstitial can be assigned to substages below 20°K in both Ta and Nb, while the $\langle 110 \rangle$ split interstitial is thought to be migrating at the more elevated temperatures of 95°K in Nb and 170°K in Ta. (Dissert. Abstr.)

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43320 (GEMP-573 (rev.)) EVALUATION OF THE POTENTIAL OF SELECTED ALLOYS FOR USE AS A FUEL CLADDING MATERIAL IN AN LMFBR. Collins, C. G.; Moteff, J.; Chandler, B. A. (General Electric Co., Cincinnati, Ohio, Missile and Space Div.). Initially Issued Nov. 1967; Final Issue June 1969. Contract AT(40-1)-2847. 312p. Dep. CFSTI.

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, 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-Ni alloys and that those with development potential offering higher creep strengths include the nickel alloys with less than ~50 percent nickel, and the cobalt, vanadium, molybdenum, and niobium alloys. The ferritic steels, nickel alloys with greater than ~50 percent nickel, and the 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 include 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)

50977 (BNWL-932) IRRADIATION EFFECTS ON REACTOR STRUCTURAL MATERIALS. Quarterly Progress Report, August-October 1968. Claudson, T. T. (ed.) (Battelle-Northwest, Richland, Wash. Pacific Northwest Lab.). Dec. 1968. Contract AT(45-1)-1830. 234p. Dep. CFSTI.

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 of, at 300 and 400°C

50980 (AECL-3365) CURRENT KNOWLEDGE OF ZIRCONIUM ALLOYS FOR REACTOR USAGE. Boulton, J. (Atomic Energy of Canada Ltd., Pinawa (Manitoba). Whiteshell Nuclear Research Establishment). June 1969. 18p. (CONF-690615-5). Dep. CFSTI (U. S. Sales Only). CAN \$0.50.

From 9th Annual Nuclear Association Conference, Montreal, Canada.

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-2½% 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-2½% 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-2½% 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)

51088 MEASUREMENT OF STORED-ENERGY RELEASE AND SPECIFIC HEAT OF NEUTRON-IRRADIATED METALS BY MEANS OF A DIFFERENTIAL HEAT FLUX CALORIMETER. Rau, F. (Institut fuer Festkoerper- und Neutronenphysik, Juelich, Ger.); Wenzl, H. Z. Angew. Phys., 27: 346-54 (Aug. 27, 1969). (In German).

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 (4.6°K). The calorimeter can be used at constant or linearly increasing temperature in the range from 4°K to 300°K. 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 1.5×10^{-5} W for $T \approx 270^\circ\text{K}$ to 4×10^{-4} W for $T \approx 4^\circ\text{K}$. For a heat release of 2 mW, the total experimental error is $\leq 9\%$ at $T \geq 30^\circ\text{K}$ and $\leq 18\%$ at $T \leq 20^\circ\text{K}$. Measurements of stored energy release in niobium and copper after 4.6°K neutron irradiation are given as examples. (auth)