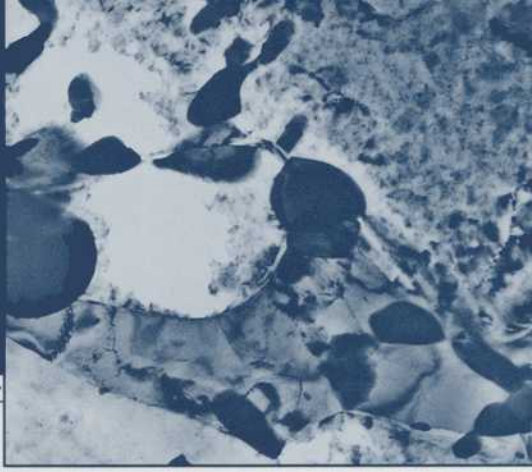

Effects of RADIATION on Materials

15TH INTERNATIONAL SYMPOSIUM

STOLLER, KUMAR, GELLES

EDITORS

STP
1125

ASTM

STP 1125

Effects of Radiation on Materials: 15th International Symposium

Roger E. Stoller, Arvind S. Kumar, and David S. Gelles, editors

ASTM Publication Code No. (PCN)
04-011250-35

ASTM
1916 Race St.
Philadelphia, PA 19103

ASTM Publication Code Number (PCN): 04-011250-35
ISBN: 0-8031-1477-X
ISSN: 1050-7515

Peer Review Policy

Each paper published in this volume was evaluated by three peer reviewers. The authors addressed all of the reviewers' comments to the satisfaction of both the technical editor(s) and the ASTM Committee on Publications.

The quality of the papers in this publication reflects not only the obvious efforts of the authors and the technical editor(s), but also the work of these peer reviewers. The ASTM Committee on Publications acknowledges with appreciation their dedication and contribution to time and effort on behalf of ASTM.

Printed in Ann Arbor, MI
August 1992

Foreword

This publication, *Effects of Radiation on Materials: 15th International Symposium,* contains papers presented at the 15th International Symposium on the Effects of Radiation on Materials, which was held in Nashville, Tennessee, 19–21 June 1990. The symposium was sponsored by ASTM Committee E-10 on Nuclear Technology and Applications. Roger Stoller, Oak Ridge National Laboratory, was the symposium chairman. Arvind S. Kumar, University of Missouri-Rolla, and David S. Gelles, Battelle Pacific Northwest Laboratory, served as cochairmen.

Attendees at the 15th International Symposium on the Effects of Radium on Materials gather outside the Doubletree Hotel in Nashville, Tennessee, for a group photo.

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Overview

ASTM Committee E-10 on Nuclear Technology and Applications sponsors a biennial series of symposia on the effects of radiation on materials. The first symposium was held in 1960 and followed an earlier series begun in 1956 by E-10, then called the Committee on Radioisotopes and Radiation Effects. Since that first meeting, these symposia have continued to grow in importance as nuclear energy has provided an increasingly larger fraction of the world's electrical capacity. The meetings have become a major international forum for the presentation and discussion of research on the influence of radiation on the microstructure and mechanical properties of structural materials. The Proceedings of the Fifteenth International Symposium on the Effects of Radiation on Materials are published in this ASTM Special Technical Publication (STP) 1125. The Symposium was held in Nashville, Tennessee from 19 to 21 June 1990. In conjunction with the Symposium, an ASTM Standards Technology Training Course on Condition Assessment and Surveillance of Nuclear Reactor Pressure Vessel Steels was held on June 18.

The Fifteenth Symposium marked a milestone in international representation. It was the first symposium in this series to have more than one half of its papers presented by authors from outside the United States. Out of the 91 papers in this book, only 35 had primary authors from the United States. In addition, roughly 45% of the symposium registrants were from institutions located outside the United States. A high level of international collaboration can also be observed in the joint, multinational authorship of many of the papers. This cooperation has in part sprung from the many informal discussions that have taken place at earlier symposia. Thus, the broad international participation and collaboration are a reflection of the fact that materials problems do not respect national boundaries and emphasizes the value of such meetings to provide an opportunity to seek common solutions to these problems.

The 91 papers presented in this STP are arranged in seventeen sections that follow the general outline of the Symposium. The first section is comprised of two papers that describe irradiation facilities. A high-temperature irradiation capsule designed for use in the Fast Flux Test Facility (FFTF) is discussed in one of these papers. Potential applications for these capsules include irradiating fuels for advanced fission reactors and ceramic components for fusion. The second paper describes a set of neutronics calculations for a special-purpose neutron source that could be used to investigate the influence of very high-energy neutrons.

The papers in the next four sections are focused on current commercial power reactor systems. Three of the sections are concerned primarily with steels, and the fourth with zirconium alloys. The five papers in the first of these sections discuss data from operating reactor surveillance programs in several countries. Four of the papers are concerned with light water reactor pressure vessel embrittlement, while one paper reports on Nimonic PE16 tie bars used in advanced gas-cooled reactors. The research discussed in this section is relevant to issues such as nuclear plant life extension and the influence of composition on embrittlement. The need to extend and supplement existing surveillance programs provides some impetus for the research discussed in the last paper, which describes the fabrication and testing of reconstituted Charpy specimens. Both insert size and welding technique are examined.

Microstructural characterization of irradiated pressure vessel steels is the topic of the six papers in the next section. The work reported here includes the examination of both engineering and simple model materials. The major techniques used were transmission electron microscopy (TEM), small-angle neutron scattering (SANS), and positron annihilation. Information is presented on materials examined prior to and following irradiation, as well as the results of postirradiation annealing studies. Correlation of the microstructure with the observed mechanical properties is a major focus of this work. The six papers in the final pressure vessel section deal with the effect of radiation on fracture toughness. In particular, several of these papers address the potential change in shape, as well as the temperature shift, of fracture toughness curves. The relationship between the temperature shift measured from a Charpy curve and the shift in the fracture toughness curve is also discussed.

The major uses of zirconium alloys in current power reactors are fuel cladding in light water reactors and pressure tubes in CANDU-type, heavy water reactors. The fracture and tensile properties of neutron-irradiated zirconium alloys are reported in three of the papers in this section. The first two papers describe the results of testing specimens fabricated from reactor components in service for up to ten years. Although a loss of ductility and fracture toughness are observed, postirradiation annealing led to substantial recovery of the mechanical properties. The last paper discusses proton-irradiated Zircaloy-2 and the influence of precipitation on corrosion resistance in this alloy. An additional paper describing precipitation in Zircaloy-2 under proton irradiation (Motta et al.) can be found in the section entitled Radiation-Induced Solute Segregation.

The next two sections deal with Radiation Damage Fundamentals, both Primary Damage Production and Formation of the Irradiated Microstructure. Three of the papers in the first section report on the examination of high-energy displacement cascades. The cascades were created by either 14-MeV neutron or charged particle irradiation. The influence of irradiation temperature and dose were explored. Interesting observations include athermal dislocation loop formation in copper and nickel and the influence of the replacement-to-displacement ratio on the order-disorder transformation in copper-palladium. The seven papers in the second section deal with the microstructure that evolves at relatively low doses. These papers are concerned with interstitial and vacancy loop nucleation, bubble formation, and the details of point-defect trapping by solutes and preexisting microstructural features. The final paper discusses the influence of either residual or transmutant gases on crack propagation.

Phenomena that occur at higher doses are the subject of the next section. Theoretical models of irradiation creep and void swelling are presented in seven papers. Three of the papers supplement our understanding of irradiation creep by examining previously unexplored mechanisms. Higher-than-expected irradiation creep has been observed at low temperatures, and it was demonstrated that a model of creep due to transient interstitial absorption could explain the observations. The fact that dislocation glide could give rise to local point-defect transients was shown to provide another possible mechanism of creep. Other authors examined the details of point-defect reactions with the dislocation core to explain the dependence of orientation on the dislocation climb velocity and the dose dependence of creep. The conventionally used rate theory of void swelling is examined and extended in the remaining papers in this section. The formation of self-organized or modulated microstructures is also investigated.

Radiation-induced solute segregation is another broadly observed phenomenon. Solute segregation is frequently correlated with void swelling, and it may promote intergranular stress corrosion cracking (IASCC). Seven of the eight papers in this section examine solute segregation in a range of materials: austenitic and ferritic steels, Nimonic PE16, Zircaloy-2, and aluminum. Models based on the inverse Kirkendall effect were generally shown to explain the segregation behavior of major alloying elements (Cr, Mo, Ni, Si) in the steels. A model of anodic dissolution of metal from crack tips was used to demonstrate that chromium segregation could promote crack initiation during IASCC. The final paper is a theoretical treatment of the influence of solute segregation on the precipitation of second phases in binary alloys.

Interest in radiation damage in ceramics has increased recently, partly as a result of the need for ceramic components to operate in high radiation fields in fusion reactors. The section on

Ceramics and Nuclear Fuels has five papers that reflect this interest. The properties of interest include: electrical conductivity for insulators, thermal conductivity for window materials, and mechanical stability for structural applications. The recent observations of radiation-induced conductivity in alumina may have significant implications for the design of fusion reactor insulators.

The demands of fusion reactors also fuel an interest in the materials discussed in the next three sections. The topics of these sections are: Copper and Copper Alloys, Vanadium-Based Alloys, and the Properties of Low-Activation Steels. Irradiation-produced microstructures and the relationship between microstructure and mechanical properties are emphasized. The materials discussed in the first section include pure copper, commercial precipitate-strengthened copper alloys, and various dispersion-strengthened alloys. Resistance to void swelling is required for these alloys to be successfully used in fusion reactors; therefore, the formation of helium bubbles and voids following either neutron irradiation or helium implantation is reported. The high level of retained oxygen in alloys strengthened by oxide dispersions was shown to enhance void swelling, and high levels of swelling reportedly were associated with unusual tensile behavior.

One of the attractive aspects of vanadium alloys for fusion applications is that neutron irradiation leads to fairly low levels of activation products with long half-lives. However, these materials are susceptible to radiation-induced embrittlement and swelling. A variety of vanadium-based alloys have been developed to mitigate these effects. Three of the papers in this section provide the results of extensive neutron irradiation experiments to investigate these alloys, the fourth discusses a helium effects experiment using the "tritium trick." High levels of swelling (up to 93%) were observed in one alloy after irradiation to 30 dpa. Tension tests revealed hardening and a loss of ductility in essentially all the alloys. In some cases, the authors believe that the loss of ductility is primarily due to helium embrittlement, while precipitates are implicated by other authors.

In order to minimize the problem of dealing with long-lived activation products, a number of so-called low-activation steels have been developed. These materials are discussed in the four papers in Properties of Low-Activation Steels. Typically, lower activation is achieved by replacing such elements as nickel, molybdenum, and niobium with manganese, tungsten, and vanadium. Although the Fe-Cr-Ni system has been well investigated, the desire to replace nickel with manganese in austenitic stainless steels has required more detailed research of the Fe-Cr-Mn phase diagram. Such work is discussed in the first paper in this section. The results of neutron and charged particle irradiations are reported in the other papers. In general, the response of the low-activation variants is similar to the conventional steels. The major trade-off involved in the switch from nickel to manganese appears to be a loss of austenite stability under certain irradiation conditions.

The last four sections of this STP deal with microstructure and mechanical properties of conventional austenitic and ferritic or ferritic-martensitic steels, respectively. The effects of irradiation with neutrons, heavy charged particles, and high-energy electrons are reported. In the austenitic microstructures section, two of the papers describe the influence of nickel and phosphorus. Phosphorus appears to reduce swelling by refining the microstructure. It is proposed that the influence of nickel on swelling is related to its affect on the dislocation bias. The microstructure and mechanical properties of austenitic stainless steel fuel cladding irradiated to high doses in a fast reactor is also discussed in one of the papers included in the section on the mechanical properties of ferritic-martensitic steels (Seran et al.). An interest in fusion is reflected in two of the austenitic mechanical property papers. In the first, a miniaturized fatigue test machine is described. The successful use of such testing devices would reduce the amount of irradiated material required for testing. The need for welds in fusion reactor first walls led to an investigation of the properties of welded joints in the European reference

316LN stainless steel. Both the swelling resistance and the mechanical properties of the welds compared unfavorably with the base metal following neutron irradiation. The final paper discusses the correlation of swelling and mechanical properties in austenitic steels alloyed with boron and various rare-earth elements.

Ferritic-martensitic steels are attractive materials for incore components in fast reactors and fusion reactor first walls due to their high strength and good swelling resistance under fast neutron irradiation. The papers presented here describe the stability of the as-fabricated microstructure under irradiation and thermal aging. Although neutron or charged particle irradiation leads to microstructures that are different than those observed under thermal aging, the effects of radiation are reduced relative to those observed in austenitic steels. Postirradiation testing of tensile and Charpy specimens fabricated from Phenix fuel ducts provide mechanical property data for alloy EM10 irradiated to high doses. Little swelling was observed in this material and the degradation of mechanical properties was modest. In contrast to these results, much greater embrittlement was observed in 12Cr-1MoVNb and a 2-¼Cr-1Mo steel irradiated to low doses in the FFTF. The final paper discusses an irradiation and testing program for a high-strength, low-alloy ferritic steel intended for use as a reinforcing bar in concrete structures.

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ISBN 0-8031-1477-X