

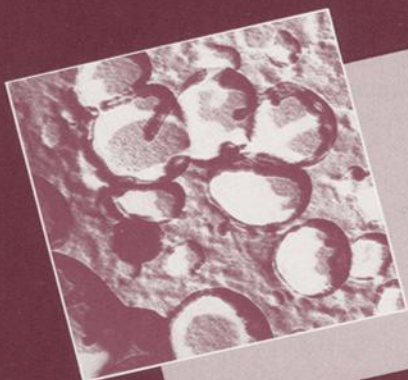
Zirconium in the Nuclear Industry

Eighth International Symposium



Van Swam/Eucken
editors

 STP 1023



STP 1023

***Zirconium in the Nuclear
Industry: Eighth International
Symposium***

Leo F. P. Van Swam and Craig M. Eucken, editors



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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 of time and effort on behalf of ASTM.

Foreword

The Eighth International Symposium on Zirconium in the Nuclear Industry was held 19–23 June 1988 at San Diego, CA. The symposium was sponsored by ASTM Committee B-10 on Reactive and Refractory Metals in cooperation with the International Atomic Energy Agency. Leo F. P. Van Swam, Advanced Nuclear Fuels Corporation, Maxi C. Noe, International Atomic Energy Agency, and Craig M. Eucken, Teledyne Wah Chang Albany, served as chairmen of the symposium. Leo F. P. Van Swam and Craig M. Eucken are editors of the resulting publication.

Contents

Overview ix

BEHAVIOR OF PRESSURE TUBES

Corrosion and Hydriding of N Reactor Pressure Tubes—DONALD D. LANNING,
A. BURT JOHNSON, JR., DENNIS J. TRIMBLE, AND S. M. BOYD 3

**Oxidation and Deuterium Uptake of Zr-2.5Nb Pressure Tubes in CANDU-PHW
Reactors**—VINCENT F. URBANIC, BRIAN D. WARR, ANGELA MANOLESCU,
C. K. CHOW, AND MICHAEL W. SHANAHAN 20

Evaluation of Zircaloy-2 Pressure Tubes from NPD—CHRISTOPHER E. COLEMAN,
BRIAN A. CHEADLE, ALLAN R. CAUSEY, PETER C. K. CHOW, PAULINE H.
DAVIES, MURRAY D. McMANUS, DOUGLAS K. RODGERS, STEFAN SAGAT, AND
GARY VAN DRUNEN 35

**Growth, Fracture, and Nondestructive Evaluation of Hydride Blisters in Zr-2.5-Nb
Pressure Tubes**—MARC LEGER, G. D. MOAN, A. C. WALLACE, AND
N. J. WATSON 50

Effects of Hydride Morphology on Zr-2.5Nb Fracture Toughness—ANDREW C.
WALLACE, GORDON K. SHEK, AND OLEV E. LEPIK 66

**Effects of Loading and Thermal Maneuvers on Delayed Hydride Cracking in
Zr-2.5-Nb Alloys**—GORDON K. SHEK AND DAVID B. GRAHAM 89

CORROSION I

**Influence of Chemical Composition on Uniform Corrosion of Zirconium-Base Alloy
in Autoclave Tests**—CRAIG M. EUCKEN, PETER T. FINDEN, SIEGRUN
TRAPP-PRITSCHING, AND HANS G. WEIDINGER 113

**Correlation Between 400°C Steam Corrosion Behavior, Heat Treatment, and
Microstructure of Zircaloy-4 Tubing**—TOMAS THORVALDSSON, THOMAS
ANDERSSON, ANDERS WILSON, AND ANTONY WARDLE 128

**Influence of the Manufacturing Process on the Corrosion Resistance of Zircaloy-4
Cladding**—JOHN J. SCHEMEL, DANIEL CHARQUET, AND JEAN-FRANCOIS
WADIER 141

A Laboratory Method to Predict Hydriding Properties of Zirconium Alloys Under Irradiation —A. BURT JOHNSON, JR., I. S. LEVY, DONALD D. LANNING, F. S. GERBER, AND DENNIS J. TRIMBLE	153
Development of a Mechanistic Model to Assess the External Corrosion of the Zircaloy Claddings in PWRs —PHILLIPPE BILLOT, PIERRE BESLU, ALPHONSE GIORDANO, AND JOEL THOMAZET	165

CORROSION II

Lithium Uptake and the Accelerated Corrosion of Zirconium Alloys —NATESAN RAMASUBRAMANIAN, NIC PRECOANIN, AND VICTOR C. LING	187
Microstructure and Corrosion Studies for Optimized PWR and BWR Zircaloy Cladding —FRIEDRICH GARZAROLLI, ECKHARD STEINBERG, AND HANS G. WEIDINGER	202
Corrosion Performance of Zircaloy-2 and Zircaloy-4 PWR Fuel Cladding —PETER RUDLING, HÅKAN PETTERSSON, THOMAS ANDERSSON, AND THOMAS THORVALDSSON	213
Development of a Cladding Alloy for High Burnup —GEORGE P. SABOL, GERALD R. KILP, MALCOLM G. BALFOUR, AND ELWYN ROBERTS	227
Enhanced Low-Temperature Oxidation of Zirconium Alloys Under Irradiation —BRIAN COX AND VILNIUS FIDLERIS	245
Internal Hydriding in Irradiated Defected Zircaloy Fuel Rods —JOHN C. CLAYTON	266

NODULAR CORROSION

A Systematic Survey of the Factors Affecting Zircaloy Nodular Corrosion —KEIZO OGATA, MISHIMA YOSHITSUGU, TADATSUNE OKUBO, TOSHIMASA AOKI, TAKUYA HATTORI, TORU FUJIBAYASHI, MASAHISA INAGAKI, KAZUO MUROTA, TATSURO KODAMA, AND KATSUHIRO ABE	291
Corrosion Performance Ranking of Zircaloy-2 for BWR Applications —PETER RUDLING AND ALBERT J. MACHIELS	315
Influence of Chemical Composition and Manufacturing Variables on Autoclave Corrosion of the Zircaloys —RONALD A. GRAHAM, JACK P. TOSDALE, AND PETER T. FINDEN	334
Effects of Alloying Element Distribution on the Nodular Corrosion of Zircaloy-2 —KEIZO OGATA	346
Electron Microscopy Study of Oxide Films Formed on Zircaloy-2 in Superheated Steam —BANG-XIN ZHOU	360
Heterogeneous Scale Growth During Steam Corrosion of Zircaloy-4 and 500°C —DANIEL CHARQUET, ROLAND TRICOT, AND JEAN-FRANCOIS WADIER	374

Automatic System for Measuring the Zirconium Liner and Zircaloy-2 Thickness of Zirconium Liner Tubes —MASAHIRO HONJI, TAKAHIDE SAKAMOTO, YOSHIFUMI MORIMOTO, AND SHIGETOSHI HYODO	392
--	-----

BASIC METALLURGY

Solubility Limits and Formation of Intermetallic Precipitates in ZrSnFeCr Alloys —DANIEL CHARQUET, ROLAND HAHN, ERHARD ORTLIEB, JEAN-PIERRE GROS, AND JEAN-FRANCOIS WADIER	405
Direct Measurement of Matrix Composition in Zircaloy-4 by Atom Probe Microanalysis —BOEL WADMAN AND HANS-OLOF ANDRÉN	423
Diffusion of 3D Transition Elements in α-Zr and Zirconium Alloys —GAVIN M. HOOD AND RAYMOND J. SCHULTZ	435
Beta-Quenched Zircaloy-4: Effects of Thermal Aging and Neutron Irradiation —WALTER J. S. YANG AND RONALD B. ADAMSON	451
Intermetallic Precipitates in Zirconium-Niobium Alloys —XIANYING MENG AND DEREK O. NORTHWOOD	478
Magnetic Study of Zircaloy —CHIE MIYAKE AND TOSHITO TAKAMIYA	487
Oxidation Kinetics and Auger Microprobe Analysis of Some Oxidized Zirconium Alloys —ROBERT A. PLOC	498

MECHANICAL BEHAVIOR, STRESS CORROSION, AND FATIGUE

Simulated Fuel Expansion Testing of Zircaloy Tubing —JOHN P. FOSTER AND R. A. LEASURE	517
Link Between Stress Corrosion and Corrosion-Fatigue Behavior of Zircaloy in an Iodine Environment —ISABELLE SCHUSTER AND CLÉMENT LEMAIGNAN	535
Effects of Irradiation and Hydriding on the Mechanical Properties of Zircaloy-4 at High Fluence —ANAND M. GARDE	548
Applications of Crystallographic Textures of Zirconium Alloys in the Nuclear Industry —K. LINGA MURTY	570
Stress Versus Strain Rate Characteristics of Zircaloy Cladding Tubes Subjected to Various Deformation Paths —HIROSHI SUZUKI AND TADATSUNE OKUBO	596

CREEP AND GROWTH

The Influence of Tin Content on the Thermal Creep of Zircaloy-4 —WILLIAM A. MCINTEER, DAVID L. BATY, AND KIM O. STEIN	621
--	-----

Effects of High Neutron Fluences on Microstructure and Growth of Zircaloy-4 —FRIEDRICH GARZAROLLI, PETER DEWES, GERD MAUSSNER, AND HANS-HENNING BASSO	641
Accelerated Irradiation Growth of Zirconium Alloys —MALCOLM GRIFFITHS, ROSS W. GILBERT, AND VILIUS FIDLERIS	658
Creep Behavior of Zircaloy Cladding Under Variable Conditions —YUTAKA MATSUO	678
Effects of Microstructural Factors on Irradiation Growth in Zirconium-Niobium Alloys —NORIYUKI SADAOKA AND MOTOMASA FUSE	692
The Effect of Temperature on the Irradiation Growth of Cold-Worked Zr-2.5 Nb —RICHARD A. HOLT AND RONALD G. FLECK	705
Discussion Section	725
Index	763

Overview

This volume contains papers presented at the Eighth International Symposium on Zirconium in the Nuclear Industry held in San Diego, CA, in June 1988. Zirconium alloys have found widespread application in the core of nuclear power plants since their first use in the early 1950s. The behavior of zirconium alloys in reactor environments and in laboratory tests designed to simulate some aspect of in-reactor service conditions has been of ongoing interest since the first application of zirconium to reactor usage. The symposia were organized to provide a forum for technical discussion of zirconium alloy applications and properties. The first of these meetings was held in Philadelphia in 1968, and the published proceedings of this and subsequent conferences have become one of the most significant bodies of information on zirconium alloy behavior available in the literature. Most of the papers presented at the San Diego symposium are published here after careful peer review and editing. The most significant parts of the discussion that ensued after the oral presentation of each paper are included as well.

Previous symposia in this series have placed particular emphasis on the use of zirconium alloys, especially the Zircalloys (Zr-1.5Sn alloys with small additions of iron, chromium, and nickel), in light water reactors. A special session of this symposium was devoted to behavior of pressure tubes, especially the Zr-2.5Nb pressure tubes used in the Canadian Deuterium Uranium (CANDU) pressurized heavy water reactors.

While the Zircalloys have proven to be quite successful in their designed usage, a desire for longer lifetimes of core components and increased duty cycle puts more demand on materials performance. This demand has led to more in-depth studies of phenomena associated with zirconium alloy corrosion mechanisms, to fine tuning of the Zircaloy composition and to evaluation of new zirconium alloys. In-reactor experience formed the basis for about a third of the papers presented. New data for many of the topics covered in previous symposia are presented, including irradiation growth, stress corrosion cracking, hydriding, texture and mechanical properties, and creep and corrosion behavior. Increased emphasis on understanding the basis of the kinds of behavior observed resulted in a number of papers dealing with compositional or structural phenomena occurring on a microscopic, or even on an atomic scale.

Several presentations described the importance of iron content or distribution on corrosion resistance of Zircaloy. Bulk iron content, iron content within the matrix zirconium as measured by atom probe, and variability of iron content within 10 μm areas were reported to be related to either uniform or nodular corrosion resistance of Zircaloy. High or uniform iron content was found to be beneficial. Diffusion of iron in zirconium was reported to be extremely rapid at temperatures as low as 500°C. The solubility of iron in zirconium at 820°C was reported to be 120 ppm. Intermetallic particles of zirconium-iron-chromium were ob-

served to precipitate when Zircaloy-4 is quenched from the beta phase. The size and distribution of these Laves phase intermetallic particles is thought to influence the corrosion behavior of the Zircalloys.

Papers dealing with nodular corrosion of Zircaloy once again were a significant part of the symposium. Processing variables are known to have an impact on nodular corrosion resistance, and processes that are characterized by low-temperature deformation and annealing were reported to result in better nodular corrosion resistance than higher temperature processing. Intermetallic particle distribution has previously been reported as affecting nodular corrosion resistance, and work presented in this conference showed that the presence of large variations in the content of the intermetallic particle forming elements iron, chromium, and nickel leads to degradation of nodular corrosion resistance. Out of reactor tests to evaluate the nodular corrosion resistance of zirconium alloys have been used for several years, and a test method involving autoclave exposure of samples to steam at 520°C was proposed as a more representative test for discriminating nodular corrosion resistance than these previous tests.

The effect of annealing on the corrosion resistance of the Zircalloys was described by several authors. A specific composition within the Zircaloy-2 chemistry range was shown to be insensitive to the detrimental effect of accumulated annealing on nodular corrosion resistance. A specific range of total annealing, as expressed by an annealing parameter, was recommended for best uniform corrosion resistance of Zircaloy-4 in pressurized water reactors (PWRs) and for nodular corrosion of Zircaloy-4 in BWR's, but another author reported that the achievement of optimum uniform corrosion resistance must include not only a total annealing range but also specific parameters of the final anneal.

The proceedings also include two papers, which have historical as well as technical significance to the use zirconium alloys in the nuclear industry. The W. J. Kroll Zirconium Medal for 1986 and 1987 were presented to Dr. J. Googin and Dr. Y. Mishima, respectively. The text of the award recipients' speeches are included in this volume as the first two papers.

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