24TH SYMPOSIUM ON EFFECTS OF RADIATION ON NUCLEAR MATERIALS AND THE NUCLEAR FUEL CYCLE

Sponsored by ASTM Committee E10 on Nuclear Technology and Applications and ASTM Committee C26 on Nuclear Fuel Cycle

June 24-26, 2008
Sheraton Denver Hotel
Denver, Colorado

Symposium Co-Chairmen: Jeremy Busby
Oak Ridge National Laboratory
Oak Ridge, Tennessee
USA

Brady Hanson
Pacific Northwest National Laboratory
Richland, Washington
USA

Takuya Yamamoto
University of California-Santa Barbara
Santa Barbara, California
USA

MONDAY, JUNE 23, 2008

6:00 PM - 7:00 PM Welcome Reception (Cash Bar)

TUESDAY, JUNE 24, 2008

8:20 AM
Welcome Remarks
J. Busby, Symposium Co-Chairman

OPENING PLENARY SESSION

8:30 AM
Plenary Talk: A Perspective on Our Field of Study
M. Mitchell, Division of Component Integrity, US Nuclear Regulatory Commission
SESSION 1: REACTOR PRESSURE VESSEL-I

Session Chairs: R.K. Nanstad, ORNL and T. Yamamoto, UCSB

9:10 AM  
Flux Effects on Microstructures, Hardening and Embrittlement of Irradiated RPV Steels: A High Flux–High Fluence Model  
G. R. Odette, T. Yamamoto, University of California Santa Barbara, Santa Barbara, California, USA;  
R. Stoller and R. K. Nanstad, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA;  
M. EricksonKirk, US Regulatory Commission, Rockville, Maryland, USA;  
E. V. Mader, Electric Power Research Institute, Idaho Falls, Idaho, USA

9:40 AM  
BREAK

10:00 AM  
A Review of High Fluence Data, and Assessment of its Applicability to the Prediction of Embrittlement Trends for Extended Reactor Lifetimes, and for New Reactors  
M. EricksonKirk, NRC, Sykeville, Maryland, USA

10:20 AM  
International Atomic Energy Agency (IAEA) Coordinated Research Projects on Structural Integrity of Reactor Pressure Vessels  
W. L. Server, ATI Consulting, Pinehurst, North Carolina, USA; and R. K. Nanstad, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA

11:00 AM  
Embrittlement Correlation Method for the Japanese Reactor Pressure Vessel Materials  
N. Soneda, K. Doshi, A. Nomoto, K. Nishida, S. Ishino, Central Research Institute of Electric Power Industry, Tokyo, Japan

11:20 AM  
ASTM – My Old Friend  
L. Hiu, Shanghai, Peoples Republic of China

11:35 AM  
Development of Small Specimen Test Technique (SSTT) for its Eventual Standardization in the IFMIF/EVEDA Activity  
H. Matsui, E. Wakai, and H. Tanigawa, Kyoto University, Uji, Kyoto, Japan

11:50 AM  
BUFFET LUNCH  (Hosted)
SESSION 2: REACTOR PRESSURE VESSEL-II

Session Chairs: B. Odette, UCSB, and W. Server, ATI Consulting

1:00 PM
The Effect of Post-Irradiation Annealing on VVER-440 RPV Materials Mechanical Properties and Nano-structure under Re-irradiation
A. Chernobaeva, D. Y. Erak, and O. Zabusov, RRC Kurchatov Institute, Moscow, Russia; S. Rogozkin, ITEP, Moscow, Russia; and L. Debarberis, JRC EC, The Netherlands

1:20 PM
Nanocluster Formation and Dissolution in VVER-1000 RPV Steels during Neutron Irradiation and Post Irradiation Annealing
M. K. Miller, K. F. Russell, and R. K. Nanstad, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA; A. A. Chernobaeva, Ya Shtrombakh, D. Erak, and O. Zabusov, Kurchatov Institute, Moscow, Russia

1:40 PM
The Fast Neutron Flux Effect on Radiation Embrittlement of VVER-440 RPV Steels
Y. A. Nikolaev, Y. N. Korolev, and I. A. Gaisin, Kurchatov Institute, Moscow, Russia

2:00 PM
The Effect of Initial Irradiation and Chemical Composition on Re-Irradiation Embrittlement of Cr-Mo VVER-440 RPV Steels
Y. A. Nikolaev, Kurchatov Institute, Moscow, Russia

2:20 PM
Post Mortem Investigations of Greifswald WWER-440 Reactor Pressure Vessels
H. W. Viehrig, U. Rindelhart, and J. Schuhknecht, FZR, Rossendorf, Dresden, Germany

2:40 PM  BREAK

SESSION 3: RADIATION DAMAGE I

Session Chairs: S. Maloy, LANL, and T. S. Byun, ORNL

3:00 PM
Swelling and Creep Observed in AISI 304 Fuel Pin Cladding From Three MOX Assemblies Irradiated In EBR-II
F. Garner, Pacific Northwest National Laboratory, Richland, Washington, USA; B. J. Makenas, and A. Chastain, Flour Hanford Corp., Richland, Washington, USA
24th Symposium on Effects of Radiation on Nuclear Materials and the Nuclear Fuel Cycle  
Tuesday, June 24, 2008, Session 3, continued

3:20 PM  
**Swelling-Induced Distortion of EBR-II Hexagonal Wrappers in Response to Gradients in Neutron Fluence and Irradiation Temperature**  
F. Garner, Pacific Northwest National Laboratory, Richland, Washington, USA

3:40 PM  
**Analysis of Radiation Induced Segregation and Microstructure in Neutron-Irradiated Japanese Prime Candidate Alloy**  
K. Prater and G. Was, University of Michigan, Ann Arbor, Michigan, USA; and J. T. Busby, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA

4:00 PM  
**Study of Microstructure and Property Changes in Irradiated Ass316 Wrapper of Fast Breeder Test Reactor**  
S. Saroja, C. N. Venkiteswaran, V. Karthik, P. Parameswaran, N. G. Muralidharan, V. A. Raj, V. Venugopal, M. Vijayalakshmi, K.V. Kasi Viswanathan and B. Raj, Indira Ghandhi Center for Atomic Research, Tamil Nadu, India

4:20 PM  
**Effects of High Temperature Irradiation on Selected Gen IV Structural Metallic Alloys**  
R. Nanstad and D. Hoelzer, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA; and D. McClintock, University of Texas at Austin, Austin, Texas, USA

4:40 PM  
**Radiation Effects in Cast Stainless Steels for Fusion Reactor Application**  
J. Busby, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA

5:00 PM  
**Closing Remarks**  
J. Busby, Symposium Co-Chairman

5:05 PM SYMPOSIUM ADJOURNS FOR THE DAY

**WEDNESDAY, JUNE 25, 2008**

**SESSION 4: RADIATION DAMAGE II**

**Session Chairs:** J. Busby, ORNL and M. Gusev, NPI

8:15 AM  
**Opening Remarks**  
J. Busby, Symposium Co-Chairman
24th Symposium on Effects of Radiation on Nuclear Materials and the Nuclear Fuel Cycle
Wednesday, June 25, 2008, Session 4, continued

8:20 AM
Irradiated Fracture Toughness of an Advanced Nano-Structured Ferritic Alloy and an Optimized Oxide Dispersion Strengthened Ferritic Alloy
D. A. McClintock, M. Sokolov, D. T. Hoelzer, and R. K. Nanstad, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA

8:40 AM
Irradiation-induced Hardening and Embrittlement of ODS Ferritic Steels
J. H. Lee, Kyoto University, Uji, Kyoto, Japan; R. Kasada and A. Kimura, Institute of Advanced Energy, Kyoto University, Uji, Kyoto, Japan; S. J. Oh, and C. H. Jang, KAIST, Daejeon, Republic of Korea

9:00 AM
On the Effect of Ball Milling Variables on the Formation of Nanofeatures in Nanostructured Ferritic Alloys
N. Cunningham and G. R. Odette, University of California Santa Barbara, Santa Barbara, California, USA

9:20 AM
Properties of Irradiated Nanocarbon Fibre Reinforced Ceramics
A. Horvath, F. Gillemont and M. Horvath, Atomic Energy Research Institute, Budapest, Hungary

9:40 AM BREAK

10:00 AM
Irradiation Hardening Behavior of RPV Steel and Fe-Mn Alloys after Fe-ion Irradiations
R. Kasada, H. Kishimoto, and A. Kimura, Institute of Advanced Energy, Kyoto University, Gokasho, Uji, Kyoto, Japan; H. Yano, K. Yabuuchi, Graduate School of Energy Science, Kyoto University Gokasho, Uji, Kyoto, Japan

10:20 AM
Structural Design and Analysis of the Ceramic Internals of HTR-PM
Z. Zhang, Institute of Nuclear and New Energy Technology, Tsinghua University of Beijing, China

SESSION 5: REACTOR PRESSER VESSEL-III

Session Chairs: M. Miller, ORNL and TBD

10:40 AM
Further Results on Attenuation of Neutron Embrittlement Effects in a Simulated RPV Wall
W. Server, ATI, Consulting, Pinehurst, North Carolina, USA; M. Brumovsky and M. Kytka, Nuclear Research Institute, Rez, Czech Republic
24th Symposium on Effects of Radiation on Nuclear Materials and the Nuclear Fuel Cycle
Wednesday, June 25, 2008, Session 5, continued

11:00 AM
The Role of Irradiated Properties of WWER RPV Cladding at PTS
F. Gillemot, Atomic Energy Research Institute, Budapest, Hungary

11:20 AM
Attenuation of Radiation Damage through RPV Wall
M. Brumovsky, Nuclear Research Institute, Rez, Czech Republic

11:40 AM
Variation of Properties through Unirradiated VVER-1000 RPV Forgings and Weld Seams
Y. A. Nikolaev and A. Kostromin, Kurchatov Institute, Moscow, Russia

12:00 N BUFFET LUNCH (Hosted)

SESSION 6: RADIATION DAMAGE III-He

Session Chairs: J. Busby and B. Hanson, Symposium Co-Chairs

1:00 PM
Radiation Damage in Polymeric Materials
K. J. Leonard, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA; and E. Shin, NASA-Glenn Research Center, Cleveland, Ohio, USA

1:40 PM
In-Situ Helium Implantation Studies of High He/dpa Ratio Effects on the Cavity Evolution in Tempered Martensitic and Nanostructured Ferritic Alloys
G. R. Odette, T. Yamamoto, P. Miao, University of California Santa Barbara, Santa Barbara, California, USA; R. J. Kurtz and D. J. Edwards, Pacific Northwest Laboratory, Richland, Washington, USA; H. Tanigawa, Japan Atomic Energy Agency, Tokai-mura, Ibaraki, Japan

2:00 PM
Recent Progress on Modeling Helium Transport and Fate in Tempered Martensitic and Nanostructured Ferritic Alloys
T. Yamamoto and G. R. Odette, University of California Santa Barbara, Santa Barbara, California, USA; R. J. Kurtz, Pacific Northwest National Laboratory, Richland, Washington, USA; B. Wirth, University of California Berkeley, Berkeley, California, USA

2:20 PM
Kinetic Monte Carlo Simulation of Helium Bubble Evolution in ODS Steels
S. Sharafat and N. Ghoniem, University of California Los Angeles, California, USA; and K. Nagasawa and A. Takahashi, Tokyo University of Science, Chiba, Japan

2:40 PM BREAK
SESSION 7: RADIATION DAMAGE IV

Session Chairs: R. Stoller, ORNL and Y. Osetskiy, ORNL

3:00 PM
Irradiation Hardening and Softening Behavior in Metallic Materials
T. S. Byun and K. Farrell, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA; M. Li, Argonne National Laboratory, Argonne, Illinois, USA

3:20 PM
Interrelationship between True Stress-True Strain and Microstructures on Plastic Deformation Behavior in Neutron-Irradiated or Warm-Rolled Austenitic Stainless Steels
K. Kondo, Y. Miwa, T. Tsukada, S. Yamashita, and K. Nishoniri, Japan Atomic Energy Agency, Ibarki-Ken, Japan

3:40 PM
Influence of Neutron Irradiation on Energy Accumulation and Dissipation during Plastic Flow and Hardening of Metallic Polycrystals
O. P. Maksimkin, M. N. Gusev, and D. A. Toktogulova, Institute of Nuclear Physics, Almaty, Kazakhstan; F. A. Garner, Pacific Northwest National Laboratory, Richland, Washington, USA

4:00 PM
Investigation of Deformation Processes and Concurrent $\gamma \rightarrow \alpha$ Transformation Associated with the “Deformation Wave” Phenomenon Observed in 12Cr18Ni10Ti after Irradiation in BN-350
M. N. Gusev, I. S. Osipov, N. S. Silniagina, and O. P. Masimkin, Institute of Nuclear Physics, Almaty, Kazakhstan; F. A. Garner, Pacific Northwest National Laboratory, Richland, Washington, USA

4:20 PM
Irradiation Creep of BCC Structural Alloys
M. Li, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA

4:40 PM
Plastic-Viscoplastic Constitutive Models of Nanostructured Ferritic Alloys from 196 to 1000°C
M. Salston and G. R. Odette, University of California Santa Barbara, Santa Barbara, California, USA

5:00 PM
Closing Remarks

5:05 PM
SYMPOSIUM ADJOURNS FOR THE DAY

6:30PM
RECEPTION (Cash Bar)

7:30 PM
BANQUET
THURSDAY, JUNE 26, 2008

SESSION 8: MODELING/FUELS

8:15 AM
Opening Remarks
J. Busby, Symposium Co-Chair

Session Chairs: J. Busby, ORNL and B. Hanson, PNNL

8:20 AM
Introduction of Nuclear Standards Development Process in China
Y. Kang, Institute for Standardization of the Nuclear Industry, Beijing, China

8:40 AM
Development of a CANDU Fuel Bundle Finite Element Model to Assess Post-Irradiation Mechanical Stress Distributions
T. J. Lampman, and A. Popescu, Nuclear Safety Solutions Ltd., Toronto, Ontario, Canada; J. Freire-Canosa, Nuclear Waste Management Organization, Toronto, Ontario, Canada

9:00 AM
Atomic-Scale Modeling of Dislocation Dynamics in Environmental of Radiation Defects
Y. N. Osetsky and R. Stoller, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA; D. J. Bacon, The University of Liverpool, Liverpool, UK

9:20 AM
A Comparison of Displacement Cascade Damage Production in Nanocrystalline and Single Crystal Iron
R. Stoller, P. J. Kamenski, and Y. N. Osetsky, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA

9:40 AM
A Multiscale Model of the Master Curve Shape and Irradiation Induced Shifts Based on Measurements of the Initiation and Arrest Toughness of Cleavage Oriented Iron Single Crystals
G. R. Odette, H. Hribernik and T. Yamamoto, University of California Santa Barbara, Santa Barbara, California, USA

10:00 AM BREAK
SESSION 9: REACTOR PRESSURE VESSEL-IV

Session Chairs: M. Sokolov, ORNL and F. Gillemot, Atomic Energy Research Institute

10:20 AM
Microstructural Characterization of Copper-Containing RPV Materials Irradiated to High Fluences
N. Soneda, K. Dohi, K. Nishida, and A. Nomoto, CRIEPI, Tokyo, Japan; M. Tomimatsu, Mitsubishi Heavy Industries, Kobe, Japan; H. Matsuzawa and T. Osaki, Japan Nuclear Energy Safety Organization, Tokyo, Japan

10:40 AM
Very Large Fluence Irradiation Embrittlement of RPV Steel
M. Brumovsky, Nuclear Research Institute, Rez, Czech Republic

11:00 AM
Determination of Point Defect Concentration in Neutron-Irradiated Steels by Experimental and Computational Study
J. Kwon, and J. H. Hong, KAERI, Daejeon, Republic of South Korea

11:20 AM
Irradiation-Induced Grain Boundary Impurities and Solutes Segregation and Effects on Ductile-to-Brittle Transition Temperature in Japanese Reactor Pressure Vessel Steels
Y. Nishiyama, M. Yamaguchi, K. Ebihara, K. Onizawa, Japan Atomic Energy Agency, Ibarki-ken, Japan; and H. Matsuzawa, Japan Nuclear Energy Safety Organization, Tokyo, Japan

11:40 AM
Effects of Irradiation on Fe-C-Met Type Model Alloys
M. Brumovsky, Nuclear Research Institute, Rez, Czech Republic

12:00 N
BUFFET LUNCH (Hosted)

SESSION 10: REACTOR PRESSURE VESSEL-V

Session Chairs: M. Erickson-Kirk, NRC and H. Watanabe, Kyushu Univ.

1:00 PM
Magnox Steel Reactor Pressure Vessel Monitoring Schemes – An Overview
M. R. Wooton, C. J. Bolton, R. Moskovic, and P. E. J. Flewitt, Magnox Electric Ltd., Oldbury-on-Severn, Bristol, UK
1:20 PM
Final Results from the CARISMA Project on Fracture Mechanical Assessments of Pre-Irradiated RPV Steels Used in German PWR
H. Hein, H. Schnabel, and E. Keim, AREVA NP GmbH, Erlangen, Germany

1:40 PM
Trend curves for IAEA Reference Steel JRQ
M. Brumovsky and M. Kytka, Nuclear Research Institute, Rez, Czech Republic

2:00 PM
Analysis of the Belgian Surveillance Fracture Toughness Database Using Conventional and Advanced Master Curve Approaches
E. Lucon and M. Scibetta, SCK-CEN, Mol, Belgium; R. Gerard, Tractabel Engineering, Woluwe, Belgium

2:20 PM
The Technical Basis for Revision 3 or Regulatory Guide 1.99, Guidelines on Predicting the Effects of Radiation Embrittlement on Reactor Pressure Vessel Steels
M. EricksonKirk, U S Nuclear Regulatory Commission, Rockville, Maryland, USA

2:40 PM
The Microstructure and Hardness Changes of Irradiated A533B Steels and Model Alloys
H. Watanabe and N. Yoshida, Kyushu University, Kasugashi, Fukuoka, Japan; Y. Kamada and S. Takahashi, Iwate University, Morioka, Japan

3:00 PM BREAK

3:20 PM
Closing Plenary Talk: Looking to the Future of Radiation Damage and Reactor Materials
S. Zinkle, Materials Science and Technology Division Director, ORNL, Oak Ridge, Tennessee, USA

4:00 PM
Closing Remarks
J. Busby, Symposium Co-Chairman

4:05 PM SYMPOSIUM ADJOURNS
A Perspective on Our Field of Study

Presentation by

Dr. Matt Mitchell
Division of Component Integrity
US Nuclear Regulatory Commission
Flux Effects on Microstructures, Hardening and Embrittlement of Irradiated RPV Steels: 
A High Flux-High Fluence Model

G. R. Odette¹, T. Yamamoto¹, R. E. Stoller², R. K. Nanstad², E. V. Mader¹³, M. EricksonKirk³

¹University of California, Santa Barbara, California, USA 
²Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA 
³U. S. Nuclear Regulatory Commission, Rockville, Maryland, USA 
⁴Electric Power Research Institute, Idaho Falls, Idaho, USA 

We discuss and analyze issues associated with the use of very high flux test reactor data to develop predictive transition temperature shift (TTS) models at high fluence where there is a paucity of low flux surveillance data. Our analysis involved development and application of a physically based three-feature TTS (3FTTS) model. In addition to contributions from stable matrix defects (SMDs) and copper rich precipitates (CRPs), the 3FTTS model treats both direct hardening and indirect sink effects of thermally unstable matrix defects (UMDs) that both form and dissolve (anneal) under irradiation, with a characteristic temperature dependent recovery time, $T_{umd}(T)$. The UMDs are not significant at low flux surveillance conditions, but are very important for high-flux test reactor irradiations. The effect of UMD sinks is modeled by an effective fluence that is used in a CRP plus UMD TTS model derived from the power reactor surveillance database. High fluxes shift these CRP plus UMD contributions to higher actual fluences compared to surveillance conditions. The direct UMD hardening is simply added to the flux-fluence-temperature dependent CRP plus UMD contributions. The sink effect of UMDs was calibrated for a single material and irradiation condition. The $T_{umd}$ were derived from previously reported low temperature post irradiation annealing data. The 3FTTS model was then fit to TTS and irradiation-hardening data for a number of individual alloys with a single material-temperature dependent fitting parameter, representing the saturated UMD TTS (or hardening) contribution, $TTS_{umd}$. The 3FTTS model rationalizes most of the trends observed in the high-flux database up to or greater than fluences of $6 \times 10^{19}$ n/cm². At low flux the 3FTTS model predictions are in generally good agreement with both the surveillance and the low to intermediate flux IVAR databases. The opportunities for further development of 3FTTS-type models are outlined.
A Review of High Fluence Data, and Assessment of its Applicability to the Prediction of Embrittlement Trends for Extended Reactor Lifetimes, and for New Reactors

Mark EricksonKirk, Senior Material Engineer, USNRC, Rockville, MD, mtk@nrc.gov

The U.S. Nuclear Regulatory Commission (NRC) has approved the extension of power reactor operating licenses from 40 to 60 years for 48 reactors. Twelve more license extension requests are currently being processed, and the NRC has been informed that the industry plans to submit 23 more requests in the coming years. Both life extension and power up-rates will increase the maximum fluence to which the reactor pressure vessel will be subjected during its licensed lifetime. Current estimates suggest that after 60 years of operation the maximum fluence on the inner diameter (ID) on many pressurized water reactors will be as high as $7-8 \times 10^{19} \text{ n/cm}^2 (E>1\text{MeV})$. Additionally, some new reactor designs have 60 year design fluences as high as $1 \times 10^{20} \text{ n/cm}^2$ on the ID. Very little of the data from existing 10 CFR Part 50 Appendix H surveillance programs have fluences of these magnitudes. Consequently, embrittlement trend curves (ETC) that are derived as fits to surveillance databases may not provide robust predictions of embrittlement in the high fluence regime.

As part of license extension plans licensees are required to have, or plan to have, surveillance data at fluences that are between one and two times the predicted peak ID fluence at 60 years. However, because of the limited lead factors characteristic of surveillance capsules, in most cases these data will not be available for some time to come. It is therefore desirable to use existing information to provide an advance indication of what these 60 year capsules may show, and to inform possible revisions to the surveillance-calibrated ETCs that are being used today. To this end we review the literature from the past three decades, with the specific goal of identifying data at high fluences (here defined as $O > 3 \times 10^{19} \text{ n/cm}^2$). These data fall into three categories:

A. Data at both power and test reactor fluxes
B. Data at power reactor fluxes only
C. Data at test reactor fluxes only

Our evaluation includes an assessment of the effect, or lack thereof, of fluence rate (flux) on the embrittlement magnitude that occurs at high fluence. Additionally these data provide an opportunity to assess how well, or not, ETCs that have been developed as statistical fits to existing surveillance data extrapolate to fluence that lie largely beyond their calibrated range.
The International Atomic Energy Agency (IAEA) has conducted a series of Coordinated Research Projects (CRPs) that have focused on irradiated reactor pressure vessel (RPV) steel fracture toughness properties and approaches for assuring structural integrity of RPVs after extended service times. A series of nine CRPs have been sponsored by the IAEA, starting in the early 1970s, focused on neutron radiation effects on RPV steels. In conjunction with the CRPs, information exchanges have included many consultants' meetings, specialists' meetings, and international conferences dating back to the mid 1960s. In 1972, twenty-five countries operated water cooled type reactors. Individual studies on the basic phenomenon of radiation hardening and embrittlement were performed in these countries to better understand increases in tensile strength and shifts to higher temperatures for the ductile-brittle transition temperature. The purpose of the CRPs was to develop correlative comparisons to test the uniformity of results through coordinated international research studies and data sharing.

Two basic mechanisms of irradiation embrittlement have been identified that result in radiation hardening and shifts in transition temperature: (1) matrix damage (MD) due to irradiation-produced point defect clusters and dislocation loops; and (2) irradiation-enhanced formation of copper-enriched clusters (CEC) in older RPV steels containing residual amounts of copper. The understanding and modeling of these mechanisms have evolved over the last forty years, and sophisticated embrittlement correlations have been developed that incorporate the knowledge of these mechanisms. Considerations of dose rate effects, effects of other alloying (nickel, manganese, silicon, etc.) and residual elements (eg., phosphorus), and drop in upper shelf toughness are also important for assessing neutron embrittlement effects.

The ultimate use of embrittlement understanding is application to assure structural integrity of the RPV under current and future operation and accident conditions. Material fracture toughness is the key ingredient needed for this assessment, and many of the CRPs have focused on measurement and application of irradiated fracture toughness. This paper presents an overview of the progress made since the inception of the CRPs in the early 1970s. The chronology and importance of each CRP will be reviewed and put into context for continued safe operation of RPVs. The importance of knowing the fracture toughness of irradiated RPV materials and application to extended life operation of the RPVs is emphasized.
Prediction of the embrittlement of the reactor pressure vessel (RPV) materials is very important for the structural integrity assessment of RPVs. The current Japanese embrittlement correlation equation JEAC4201, as of August 2007, for the transition temperature shift prediction was developed and issued in 1991, and has not been revised for more than 15 years. Recent accumulation of surveillance data as well as the improvement of the understanding on the embrittlement mechanisms have enabled us to develop a new embrittlement correlation with an appropriate mechanistic background. In this paper, we present a new embrittlement correlation method that is now under approval process, as of August 2007, as a revision of the current JEAC4201. In the project of this correlation development, we performed microstructural analyses of some of the Japanese surveillance materials primarily using three-dimensional atom probe and transmission electron microscope techniques to understand microstructural changes due to real reactor irradiation. Two important new findings in these observations are 1) solute atom cluster formation with or without copper atoms is responsible for the embrittlement of the surveillance materials, and 2) clear flux effect exists both in the mechanical property and the microstructures of the materials irradiated at different fluxes. Based on these findings together with the knowledge on the embrittlement mechanisms from the literature, we developed a new embrittlement correlation method that consists of a set of rate equations that models the microstructural changes due to irradiation and the equations that correlate the predicted microstructural changes with embrittlement. The coefficients of the equations were optimized using the latest Japanese surveillance data. The input data for this new correlation method are copper and nickel contents, irradiation temperature, neutron flux and neutron fluence. This correlation method has only one set of coefficients with which embrittlement of all the materials used for the Japanese RPVs, regardless of the material product form, can be predicted with a standard deviation of 9.4°C.
ASTM - My Old Friend

Presented by

LiHui
Shanghai
Peoples Republic of China
The Effect of Post-Irradiation Annealing on WER-440 RPV Materials Mechanical Properties and Nano-Structure Under Re-Irradiation

Authors: A. Chernobaeva¹, S. Rogozkin², D. Erak¹, O. Zabusov¹ and L. Debarberis³ - RRC KI (Russia), ²-ITEP (Russia), ³-JRC, (the Netherlands)

Introduction

Some WER-440 units in Russia and Europe are operated after annealing of reactor pressure vessel (RPV). Safety operation of these units requires reliable forecast of mechanical properties behavior under irradiation for the weld N» 4, located in front of the core. That is the element of construction, which limits radiation safe life of the whole pressure vessel.

At the present prediction of radiation embrittlement of RPV steels under re-irradiation after annealing is carried out using «lateral shift» scheme. Estimation of mechanical properties behavior under re-irradiation using «lateral shift» is characterized by over conservatism. So, it is rather actual problem to considerate new model notions.

"Primavera" is an international project concerning this problem. RRC «Kurchatov Institute)) (Russia), JRC (The Netherlands), Fortum, VTT (Finland) and ISM (Bulgaria) participate in this project.

The mechanical properties study in unirradiated, irradiated, and re-irradiated states were reported at the papers in journal "Pressure vessel and piping". The present work provides the analyses of results of mechanical properties and APT study of nano-structure evolution of WER-440 RPV materials under irradiation and re-irradiation.
The microstructures of a high nickel, low copper VVER-1000 (15Kh2NMFAA) base metal (1.34 wt% Ni, 0.47% Mn, 0.29% Si and 0.05% Cu), and a high nickel, low copper (12Kh2N2MAA) weld metal (1.77 wt% Ni, 0.74% Mn, 0.26% Si and 0.07% Cu) that were neutron irradiated to fluences of up to 14.9 x 10^17 m^-2 and up to 11.5 x 10^17 m^-2 (E > 0.5 MeV), respectively, have been characterized by atom probe tomography. The fluence dependencies of AT41J.ADJ values indicated increased radiation sensitivity of the higher nickel weld metal. High number densities of ~2-nm-diameter Ni-, Si- and Mn-enriched nanoclusters were found in the neutron irradiated base and weld metals. No copper enrichment was associated with these nanoclusters and no copper-enriched precipitates were observed. Given the local variability in these materials, there is a reasonable correlation between the observed shift in the Charpy mechanical property data with the number density of nanoclusters. These nanoclusters were present after a post irradiation anneal of 2 h at 450 °C but had dissolved into the matrix after 24 h at 450 °C. Phosphorus, nickel, silicon and to a lesser extent manganese, were found to be segregated to the dislocations.

Research at the Oak Ridge National Laboratory SHaRE User Facility was sponsored by Basic Energy Sciences, U.S. Department of Energy and by the Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission, under inter-agency agreement 1886-N695-3W and under contract DE-AC05-00OR22725 with UT-Battelle, LLC.
The Fast Neutron Flax Effect on Radiation Embrittlement of VVER-440 RPV Steels

Yury A. Nikolaev¹, Yury N. Korolev¹, and Igor A. Gaisin¹

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Abstract: Surveillance specimens in VVER-440/213 RPVs are irradiated in hermetically sealed capsules linked in a chain. There are two Charpy V-notch or Pre-cracked Charpy V-notch specimens of 10x10x55 mm or six 10-fold cylindrical tensile specimens with the gauge diameter of 3 mm in each capsule. The surveillance chains are loaded into the channels located on perimeter of the core baffle. Irradiation surveillance specimens are located on the external surface of the core baffle in front of the core. The difference of the fast neutron flux on the surveillance specimens located at the top part of a surveillance chain and its central past was found to be over 20. The difference in of the fast neutron flux on the surveillance specimens of the same surveillance chain allows to study the flux effect on radiation embrittlement of VVER-440 RPV steels. Additional data for this study can be obtained from comparison of radiation embrittlement of steels irradiated in RPVs with full core and irradiated in Rovno-1 Unit with dummy assembles on the periphery of the core. Results of VVER-440/213 RPV surveillance specimens studies and results of special research programs will be discussed. The chemical composition effect on the flux effect will be emphasized. A new model for evaluation of radiation embrittlement of VVER-440 steels will be proposed.

Keywords: reactor pressure vessel, flax effect, radiation embrittlement
The Effect of Initial Irradiation and Chemical Composition on Re-Irradiation Embrittlement of Cr-Mo WER-440 RPV Steels

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Abstract:
Four Units with VVER-440 reactor pressure vessels (RPVs) of the first generation are still in operation in Russia. All these RPVs were annealed and the lifetime of the units depends on re-irradiation embrittlement. In order to study re-irradiation embrittlement kinetics the following way is typically used. RPV steels are subjected to accelerated irradiation, then these steels are annealed and again are subjected to accelerated irradiation. In addition to the technique mentioned above some small samples (templetes) were taken from the inner side of operating RPVs to study the actual re-irradiation embrittlement of RPV steels (both weld seams and base metals). Same of those templetes were subsequently subjected to irradiation in surveillance channels of VVER-440/213 RPVs with different fast neutron fluxes. In the present study the results of the former and the latter types of re-irradiation embrittlement study are compared. The effect of fast neutron flux on re-irradiation embrittlement kinetics is emphasized. The effect of phosphorus and copper contents on re-irradiation embrittlement is discussed. A new approach to re-irradiation embrittlement model is analyzed.

Keywords: reactor pressure vessel, radiation embrittlement, ductile-to-brittle transition temperature
Nuclear power plant operators must demonstrate that the structural integrity of a nuclear reactor pressure vessel (RPV) is assured during routine operations or under postulated accident conditions. The aging of the RPV steels is monitored with surveillance program results or predicted by trend curves. Embrittlement forecast with trend curves and surveillance specimens may not reflect the reality. Accordingly, the most realistic evaluation of the toughness response of RPV material to irradiation is done directly on RPV wall samples from decommissioned NPPs. Such a unique opportunity is now offered with material from the decommissioned Greifswald NPP (WWER-440/230). The four Greifswald NPP units representing the first generation of WWER-440 reactors were shutdown in 1990 after 11-17 years of operation. RPVs in three different conditions are available:

- Unit 1 is irradiated, annealed and re-irradiated (IAI)
- Units 2 and 3 are irradiated and annealed (IA)
- Unit 4 is irradiated.

A region covering ±0.75 m above and below the core weld (SNO.1.4.) was annealed at 475°C for about 150 hours.

Trepans were taken from the reactor core belt line welding seam and base metal forging. At first the trepan of the core welding seam from Unit 1 representing the IAI condition was used for Master Curve (MC) and Charpy V-notch testing. The WWER-440 RPV welding seam is a X-butt multilayer submerged weld. It consists of a root welded with an unalloyed wire Sv-08A and filling material welded with the alloyed wire Sv-lOKhMFT. Specimens from 11 locations through the thickness of the welding seam were tested according to ASTM El 921-05. The reference temperature To was calculated with the measured fracture toughness values, Kjc, at brittle failure of the specimen. Generally, the Kjc values measured on pre-cracked and side-grooved Charpy size SE(B) specimens of the investigated weld metal follows the course of the Master Curve. The Kjc values show a remarkable scatter. More values than expected lie below the 5% fractile. Additionally, the MC SINTAP procedure was applied to determine T0SINTAP representing the brittle fraction of the data set. There are remarkable differences between To and T0SINTAP indicating macroscopically inhomogeneous welding metal. The evaluated To varies through the thickness of the trepan and, thus, the welding seam. After an initial increase of To from 10°C at the inner surface to 49°C at 22 mm distance from it, To again decreases to -41°C at a distance of 70 mm, finally increasing again to maximum 20°C towards the outer RPV wall. The lowest To value was measured in the root region of the welding seam representing a uniform fine grain ferritic structure. Beyond the welding root To shows a wavelike behaviour with a span of about 50 K. The highest To of the weld seam was not measured at the inner wall surface. This is important for the assessment of ductile-to-brittle temperatures measured on sub size Charpy specimens made of weld metal boat samples removed from the inner RPV wall. Our findings imply that these samples do not represent the most conservative condition. Nevertheless, the Charpy transition temperature TTUJ estimated with results of sub size specimens after the recovery annealing was confirmed by the testing of standard Charpy V-notch specimens.
Swelling and creep observed in AISI 304 fuel pin cladding from three MOX assemblies irradiated in EBR-II

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Void swelling of AISI 304 stainless steel is a subject of current interest to license extension of pressurized water reactors, providing an incentive to examine older experiments involving this steel.

Three 37-pin fuel assemblies designated PNL-3, PNL-4 and PNL-5 were clad with annealed 304 stainless steel. The assemblies were similar in that they contained similar fuel smear densities of mixed oxide (85-88%), plenum volumes (11.6cc) and were irradiated at similar fast fluxes (1.9, 1.7, 1.6 x 10¹⁵ E>0.1) in EBR-II Rows 3, 4 and 4 respectively, The primary difference between the three tests was the goal peak linear power level which was 6, 9.5 and 14 kW/ft respectively. These powers were achieved by varying the uranium enrichments (natural, 45%, 75% U²³⁵ in uranium in addition to the plutonium concentration that was 25wt% PuC>2 in UO₂+PUO₂ in all three tests.

Diameter changes were measured along the length of the pins and then cladding sections ~1 inch long were cut and cleaned of fuel prior to measurements of density. Usually 3 to 5 sections were cut from each fuel pin, and pins were removed at 2-3 exposure levels.

Axial and through-wall temperature gradients in the cladding increased with linear power such that PNL-5 had the highest maximum temperatures. Variations in temperature and through-wall temperature gradients were important in determining the swelling, which was also strongly sensitive to neutron flux and weakly sensitive to applied stress. Stresses gradually developed that arose from fission gas accumulation and fuel swelling to cause fuel-clad interaction. In these pins the time-averaged stress levels were rather low so that irradiation creep was a small contributor to total deformation and the "creep cessation" phenomenon was strongly operating. Swelling levels were large, however, reaching double-digit values in some cases.

The swelling data can be shown to be easily understood in terms of the interactive effects of dpa rate, temperature and temperature gradient to determine the duration of the transient regime of swelling, with stress playing a very minor role. The terminal swelling rate was ~1%/dpa at all irradiation conditions. When the swelling data are plotted vs. neutron fluence along the full axis of a given pin, the data form a loop whose width is primarily dependent on temperature and dpa rate. This loop can approach a near-zero width as the neutron flux decreases and the temperature increases. There is a lower limit of the high temperature side of the loop, however, such that the incubation intercept can not fall below -20 dpa for any irradiation condition.
Swelling-induced distortion of EBR-II hexagonal wrappers in response to gradients in neutron fluence and irradiation temperature

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Structural materials serving in reactor cores are often subject to significant gradients in neutron flux-spectra and irradiation temperature. Since dimensional changes arising from void swelling and irradiation creep are sensitive to these variables, the potential exists for reactor components to undergo significant distortion which often includes bowing that can interfere with the operation or the removal of the component.

In general, however, the fast reactor community has had very limited success in predicting gradients in creep and swelling and the resultant bowing, even when correlations have been developed to predict these distortions. It now appears that the primary difficulty in prediction of bowing arises from the fact that such correlations are usually written only in terms of neutron fluence or dpa, and do not explicitly include neutron flux or dpa rate. It is now known that both fluence and flux must be incorporated in dimensional correlations in order to adequately predict distortions. In some cases swelling can actually decrease with increasing dose when there are gradients in dose rate giving rise to the dose differences.

In order to demonstrate the necessity of incorporating flux effects into correlation development, a study published in the 1970s was revisited and reexamined. This study involved hexagonal ducts constructed from AISI 304 stainless steel that served as hexagonal wrappers in the EBR-II fast reactor core. The original authors of those studies noted that significant radial gradients in swelling were expected due to measured fluence gradients, but such gradients were not observed nor was there any significant bowing.

It is demonstrated that dose rate effects on swelling tend to cancel out effects arising from lower dose. In cases where there are no significant radial gradients in temperature such as occurs in these hexagonal ducts, the peak swelling is the same and the axial distribution of swelling is also essentially identical on all duct faces. If significant radial gradients in temperature exist the situation is somewhat more complicated.

Axial gradients in temperature do not appear to impact this phenomenon. The originally perplexing results are shown to be easily reconciled with the new understanding of flux effects on swelling. Since bowing arises from swelling-driven irradiation creep and the irradiation creep rate is proportional to the swelling rate, it now becomes possible to better predict bowing of structural components in various types of reactor concepts.
Analysis of Radiation Induced Segregation and Microstructure in Neutron-Irradiated Japanese Prime Candidate Alloy

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Advanced austenitic alloys have the potential for improved mechanical performance and irradiation tolerance over traditional 304 or 316 stainless steel. These properties may be critical for advanced reactor systems such as GenIV or GNEP concepts. One candidate alloy is the Japanese Prime Candidate Alloy (JPCA), a variant of the 15Cr-15Ni-TiC alloy family. Samples of the JPCA were neutron irradiated to different representative doses in the 1980's at the Fast Flux Test Facility. Samples were irradiated at -400 and 425°C to 27 or 40 dpa. The twenty-years since irradiation have resulted in dramatically reduced radioactivity, permitting a more detailed examination of radiation-induced changes. In this work, four JPCA specimens have been characterized. Specifically, microhardness, swelling, and radiation-induced segregation have all been analyzed. Swelling measurements on bulk specimens are in good agreement with past estimates from TEM analysis on similar specimens. Irradiation-induced hardness trends correlate well with expected behavior as a function of irradiation temperature and dose. Radiation-induced segregation has been observed in all specimens. Significant Cr depletion and Ni enrichment was measured. The characterization of the radiation-induced changes in this JPCA will also be compared to other data in the open literature on advanced austenitic alloys.
A study on mechanical properties and microstructure in an ASS316 wrapper irradiated in a 40MWt Fast Breeder Test Reactor (FBTR) has been carried out. Transmission electron microscopy studies and mechanical property evaluation were carried out on the hexagonal wrapper which has undergone different displacement damages up to a maximum of 84 dpa at an operating temperature of about 673K. Disc specimens of 8 mm diameter were extracted from different axial locations and tested using miniature specimen testing techniques. Mechanical properties were estimated using room temperature Shear Punch tests, while the microstructural changes were studied using a 200 KV Analytical TEM.

The steel irradiated to 56 dpa showed an increase in the Yield strength and ultimate tensile strength to about 850 and 1170 MPa respectively. The unirradiated steel showed a YS and UTS of 650 and 725 MPa respectively. The uniform elongation of the irradiated wrapper had also reduced to 4.46 % from a value of 28 % in the corresponding unirradiated material. Density measurements were carried out on the specimen extracted from different parts of the irradiated wrapper. The specimen subjected to an irradiation damage of 56 dpa indicated a volumetric swelling of about 2.2 %.

TEM studies showed extensive void formation at 40dpa and precipitation in addition to dislocation loops. The void density and size showed a progressive increase with displacement damage. The precipitates were identified to be only of M23C6 type. The dislocation density showed considerable reduction as compared to the unirradiated steel. The increase in strength and reduction in ductility with increase in dpa is attributed to irradiation hardening due to increase in defect density as is also seen from the dislocation substructure and increase in the void density and size. The paper would discuss the degradation in mechanical property in terms of the microstructural changes.
The performance of Generation IV reactors as a class will be determined by the behavior of advanced engineering materials. The effects of irradiation are major issues in the case of materials utilized for reactor internals and pressure vessels. The environmental conditions for most of the Gen IV reactors are generally beyond present day reactor technology, especially as regards the combinations of operating temperatures, reactor coolant characteristics, and neutron spectra. In some of the applications, the conditions lay well beyond advanced research programs in radiation effects on materials. Thus, new experimental data as well as analytical predictions of expected behavior of candidate materials at conditions for which there are no experimental data will be required.

This paper provides results for the first series of scoping irradiation experiments with selected metallic alloys, some of which are considered candidate materials for current Gen IV reactor applications, while others are considered as potential future candidate materials. The material classes represented are (1) nickel-base alloys (alloy 800H and Inconel 617; (2) advanced nano-structured ferritic alloys (14WT and 14YWT); and (3) a commercial ferritic-martensitic steel (9Cr-lMoV). Small flat tensile specimens (SS-3) were irradiated in so-called rabbit capsules in the High-Flux Isotope Reactor (HFIR) at temperatures from 550 to 700°C and to irradiation doses from about 1.28 to 1.61 dpa. Specimens have been tested in both the unirradiated and irradiated conditions. Irradiation temperatures were determined from resistivity measurements of SiC monitors placed inside each rabbit capsule.

The annealed 9Cr-lMoV shows small amounts of irradiation-induced hardening. Hardening resulting from the 580°C irradiation was significant for the Alloy 800H and Inconel 617, however, with increases in yield and ultimate strengths on the order of 50 to 100%. Results from the 700°C irradiation also show hardening, but with extremely low tensile elongations when tested at 700°C. For the ODS 14WT and 14YWT materials, the overall results do not indicate significant effects of irradiation at this relatively low irradiation exposure. Microstructural characterizations by transmission electron microscopy and atom probe tomography have been performed of some of the materials and the results will be described.

Research at Oak Ridge National Laboratory (ORNL) was sponsored by the U.S. Department of Energy Office of Nuclear Energy Science and Technology, and the ORNL SHaRE Facility, which was supported in part by the Division of Scientific User Facilities, Office of Basic Energy Sciences, U.S. Department of Energy. ORNL is managed by UT-Battelle, LLC for the U.S. Department of Energy under Contract DE-AC05-00OR22725.
Casting of austenitic stainless steels offers the possibility of directly producing large and/or relatively complex structures, such as the first wall shield modules or the divertor cassette for the ITER fusion reactor. Casting offers major cost savings when compared to fabrication via welding together quarter modules machined from large forgings. However, because of the large grain size, low dislocation density and extensive segregation of alloying elements, the strength properties of such cast components are frequently inferior to those of conventionally forged and annealed components. To improve and validate cast stainless steel as a substitute for wrought stainless steel for shield module applications, a testing program has been initiated. A key aspect of this validation in understanding irradiation effects in the cast stainless steels. Samples of cast 316L steels and wrought 316L steels have been irradiated in HFIR to 2 or 4 x 1(T n/cm² at temperatures of 90, 190, or 290°C. Both tensile specimens and fracture toughness specimens will be tested. The irradiation performance of the cast steels will be compared to the performance of wrought 316L stainless steel.
Advanced nano-structured ferritic alloys (NFAs) containing a high density of ultra-fine (2-5 nm) nanoclusters enriched in Y, Ti, and O are considered promising candidates for structural components in future nuclear systems. The microstructure of a NFA is composed of nanometer sized regions rich in Y, Ti, and O uniformly distributed in a ferritic matrix. The high number density of nanoclusters in NFAs are responsible for their superior tensile strengths compared to conventional oxide dispersion strengthened (ODS) ferritic alloys and may provide effective trapping centers for point defects and transmutation products produced during neutron irradiation. Though both conventional ODS and NFA alloys exhibit excellent high-temperature strength, low-temperature embrittlement and poor fracture toughness properties, inherent to ferritic alloys, have been a major concern for alloy designers. Advanced material processing techniques have been developed to refine the microstructure of these alloys and improve fracture toughness. Recent experimental results show NFAs can be produced that have a very low (cryogenic) ductile-to-brittle transition temperature (DBTT) while maintaining satisfactory fracture toughness in the "upper-shelf fully ductile region. Optimized thermo-mechanical processing for conventional ODS ferritic alloys also show significant improvements in fracture toughness properties while preserving exceptional high-temperature strength and ductility.

This paper summarizes the fracture toughness properties of an advanced NFA, designated 14YWT, currently being developed at Oak Ridge National Laboratory (ORNL) and an optimized conventional ODS ferritic alloy, designated Eurofer 97 ODS. For this study tensile specimens and dual-notch three-point bend bars were irradiated in small "rabbit" capsules in the High Flux Isotope Reactor at ORNL; at a 300°C target irradiation temperature and to a total dose of 1.5 displacements per atom (dpa). Fracture toughness data for the irradiated and unirradiated conditions will be presented along with supporting tensile, hardness, and microstructural characterization data. Special attention will be given to irradiation effects on fracture toughness, including radiation induced shift in DBTT and "upper-shelf degradation.

Research at Oak Ridge National Laboratory (ORNL) was sponsored by the U.S. Department of Energy Office of Fusion Energy Sciences, the U.S. Department of Energy Office of Nuclear Energy Science and Technology, and the ORNL SHaRE Facility, which was supported in part by the Division of Scientific User Facilities, Office of Basic Energy Sciences, U.S. Department of Energy. ORNL is managed by UT-Battelle, LLC for the U.S. Department of Energy under Contract DE-AC05-00OR22725.
Abstract:
Oxide dispersion strengthened (ODS) ferritic/martensitic steels have been recently considered as the fuel cladding material of a fast breeder reactor (FBR), not only because of sufficient high temperature creep strength and better swelling resistance, but because of enough resistance to neutron irradiation. On the other hand, it was confirmed that ODS ferritic steels with high chromium (Cr) content were very effective in suppressing corrosion in supercritical water and also had excellent mechanical properties at elevated temperature. In this study, the susceptibility of the high-Cr ODS ferritic steels to irradiation-induced hardening and embrittlement is investigated to make clear the dependence between Cr content and irradiation temperature.

The materials used in this research were ODS ferritic steels in which Cr content was over 14wt.%. All the specimens were irradiated with neutrons in the Japan Materials Testing Reactor (JMTR); the neutron irradiations were performed at different temperatures for several dose levels. Tensile and Charpy impact tests were carried out to study the dependence between Cr content and irradiation temperature on irradiation-induced hardening and embrittlement of ODS ferritic steels.

Preliminary results showed that at each temperature, ODS ferritic steels with different Cr contents had different results for irradiation-induced hardening and embrittlement. It means ODS ferritic steels show different irradiation characteristics for the dependence between Cr content and irradiation temperature; therefore, further irradiation experiments of ODS ferritic steels are certainly necessary to reveal the effects of Cr content and irradiation temperature on mechanical properties of the materials after neutron irradiations.
A large set of ball milling variables were systematically explored in order to find the combination(s) that lead to the most uniform distribution of nano-features in nanostructured ferritic alloys (NFAs). NFAs are dispersion strengthened by a high density of nm-scale Y-Ti-O nano-features, resulting in remarkable high temperature creep strength and radiation damage resistance for applications in advanced fission and fusion reactors. However, it is difficult to achieve a uniform distribution of nano-features. Regions of lower nano-feature concentrations are softer and experience dislocation recovery and grain growth during consolidation, leading to a bimodal grain size distribution and a reduction in the alloy's fracture toughness and strength. A screening method was developed to characterize the grain structure, and hence the nano-feature uniformity, in mechanically alloyed, annealed, and etched powders using scanning electron microscopy techniques. The milling variables were found to have a significant impact on the microstructure. Several trends for improving grain size uniformity were identified and include using smaller ball sizes, smaller initial powder sizes, lower ball mass to powder mass ratios. A final sample was produced using a combination of milling variables that had individually improved the small grain area fraction over a baseline set of milling conditions. These combined changes to the milling variables produced a sample with an improved microstructure and a small grain fraction of 0.89.
Properties of Irradiated Nanocarbon Fibre Reinforced Ceramics

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Abstract:
One of the candidate materials of the future high temperature reactors and fusion devices is the ceramics. Fusion research are mostly concentrated on the siliconcarbide based so called SiCSiC ceramics. The SiCSiC is expensive material to replace it 21 different nanocarbon fibre reinforced ceramics have been developed. These ceramics have enhanced toughness properties. 4*3 mm crosssection bars have been produced from all type and presently they are irradiated in the Budapest Research Reactor up to 3*10^19 n/cm². After irradiation the residual activity, hardness and toughness by using 3 point bend test will be measured. On the results the best composites will be selected and prepared for longer irradiation.
Irradiation hardening behavior of RPV steel and Fe-Mn alloys after Fe-ion irradiations

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To make clear irradiation hardening behavior of a reactor pressure vessel steel and Fe-based model alloys, 6.4 MeV Fe⁺-ion irradiation experiments have been systematically carried out at various sets of irradiation temperatures from room temperature to 290 °C and displacement damages from 0.01 dpa to 1 dpa, by using DuET facility in Kyoto University. The materials used in the present study were A533B (Cu~0.03wt.%), pure Fe, and Fe-xMn (x = 0.1, 0.3,1.0, 1.5) model alloys. The irradiation hardening of the materials was investigated by using a nano-indentation technique. Microstructural changes after the ion-irradiation were investigated by using FE-TEM with a FIB specimen fabrication processing.

The important notices in the experimental results are that enhancement of irradiation hardening by the Mn-addition is clearly confirmed and dose-dependence of irradiation hardening of A533B agreed exactly with that of Fe-1.5Mn. The results of microstructural observation will be reported in the symposium.
Structural Design and Analysis of the Ceramic Internals of HTR-PM

Presented by

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Further Results on Attenuation of Neutron Embrittlement Effects in a Simulated RPV Wall

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A carefully designed irradiation experiment was conducted in which a 180-mm thick reactor pressure vessel (RPV) wall has been simulated using eighteen 10-mm slices of some key RPV steels and irradiated under test reactor conditions to investigate the thorough wall attenuation of neutron embrittlement. Two of the irradiated materials were reported in 2006 (a low copper content plate and a high copper content Linde 80 flux weld). Another RPV steel, the international reference steel termed JRQ, was also extensively irradiated in the simulated RVP wall. Comparisons of predicted attenuation changes in toughness properties with measured fracture toughness and Charpy V-notch results are presented for the JRQ steel. The predictions of through-wall attenuation follow the practice defined in ASTM E 900-02 and Regulatory Guide 1.99, Revision 2, in which the attenuation of high energy neutron fluence (E > 1 MeV) is projected based upon an approximate displacements per atom (dpa) change through the wall thickness. The resultant degree of material damage using this dpa-based fluence change is estimated using current embrittlement correlation models.
The Role of Irradiated Properties of WWER RPV Cladding at PTS

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The WWER lifetime assessment guide VERLIFE and the IAEA PTS guide allow to use hypothetical underclad defects at PTS calculation if the cladding free of defects and the toughness properties are satisfying in aged properties.

Thermal ageing and irradiation effects are the main ageing mechanisms reducing the toughness of the RPV cladding.

The WWER-440 reactor has 3 layer submerged arc strip cladding. The first layer is overalloyed to balance the mixing with the base material, the second and third are made with 18/10 type titanium stabilised strip electrode.

Samples of cladding cut from the trepan of the Greifswald 8 unit material (the unit 8 Greifswald reactor was completed, but never operated, and finally dismantled) have been irradiated in the Budapest Research Reactor and tested. Tensile and hardness testing already performed on as received, irradiated, irradiated and annealed specimens. The tensile tests performed on miniature tensile bars cut from all of the three layers. Irradiation reduced the ductility and increased the strenghts, but the effect is much less than the effect of the temperature between room temperature and 300 °C.

Further study of the toughness properties are going on. The precracked Charpy size TPB specimen are already irradiated and J-R curve measurements will be performed shortly.
Results from a large scale irradiation experiment on determination of irradiation embrittlement and neutron fluence attenuation through RPV wall will be presented. Comparison with existing US NRC RG 1.99, Rev.2 is performed as well as comparison with other experimental data obtained after irradiation for similar neutron fluences in other experimental reactors. Some conservatism in assessment of neutron fluence attenuation through RPV wall in accordance with RG was found. Interesting results have been obtained for exponents in attenuation formulae as well as in neutron embrittlement formulae depending on neutron energies.
Abstract:
Distribution of properties in fourteen different unirradiated VVER-1000 reactor pressure vessel (RPV) steels was assessed. Variation of properties were studied in radial, axial and azimuth directions through the standard VVER-1000 forgings and in radial and azimuth directions through the standard VVER-1000 weld seams. Tensile and impact tests of the materials under consideration were carried out. Totally more than 50 serial curves were analyzed. The basic part of the results were obtained using VVER-1000 surveillance specimens. The current study revealed significant inhomogeneity of VVER-1000 forgings in radial and azimuth directions and VVER-1000 weld seams in radial direction. The study of VVER-1000 RPV steels inhomogeneity allowed to evaluate the safety margin for RPV lifetime evaluation.

Keywords: reactor pressure vessel steel, steel inhomogeneity, ductile-to-brittle transition temperature
Radiation Damage in Polymeric Materials

Presented by

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Fusion first wall irradiations produce ~ 2000 appm of He in Fe-based alloys at 200 dpa. Helium bubbles precipitate in the matrix as well as on dislocations, precipitate interfaces and grain boundaries (GB). Helium bubbles can act as nucleation sites for growing voids at lower temperatures and creep cavities at high temperatures. Tempered martensitic steels (TMS) are embrittled by GB He at lower irradiation temperatures, and undergo significant void swelling at high He/dpa ratios in dual ion irradiations. Nanostructured ferritic alloys (NFAs) are more resistant to irradiation damage at all temperatures, due to their high sink densities, and large number of nm-scale Y-Ti-O enriched nanofeatures (NF). At $T > 0.5T_m$, the primary potential radiation damage challenge for these alloy systems is likely to be creep embrittlement due to accumulation of GB He. The objective of this research is to characterize the transport, fate and consequences of He at fusion relevant He/dpa ratios.

An in-situ neutron irradiation He-implanter and TEM were used to characterize the microstructural evolutions in MA957, J12YWT (NFAs), Eurofer97 and F82H (TMS) in HFIR irradiations at 500°C to about 9 dpa and 380 appm He. Helium is uniformly implanted to a depth of 5 to 8 um from thin NiAl coatings. A high number density of very small ~1.3 nm He bubbles form in MA957 and J12YWT primarily located on the NF interfaces. A lower density of larger ~ 2 and 4 nm bubbles were observed in TMS Eurofer97 and F82H, respectively, along with fewer, but even larger (>10 nm) faceted cavities, that are likely voids. Boundaries in F82H are highly decorated with small He bubbles, while a boundary in MA957 was found to be relatively bubble free. These observations support the hypothesis that very high He concentrations can be managed in NFAs. These studies are continuing, including examination of additional alloys and irradiation conditions.
Recent Progress on Modeling Helium Transport and Fate in Tempered Martensitic and Nanostructured Ferritic Alloys

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We describe the development and application of a multiscale model of the transport and fate of He in irradiated nanostructured ferritic alloys (NFAs) and tempered martensitic steels (TMS) that are candidates for use in fusion first-wall and blanket structural applications. We focus on NFAs that have remarkable creep strength provided by a high density of Y-Ti-0 solute clusters and oxides nanofeatures (NF) that not only impede dislocation motion, but can also provide fine scale helium bubble nucleation sites and vacancy-interstitial recombination centers. Key characteristics of NFAs are 1) a high density (-10 m⁻³) of small (-2-4 nm diameter) NF, 2) fine to ultra-fine crystallite grain sizes and 3) high dislocation densities. The size and number density of these features can be modified by appropriate thermo-mechanical treatments. We also focus on a realistic modeling of He clustering behavior on dislocations that, as shown in the implanter experiments in a companion paper, is especially important in TMSs.

We employ molecular dynamics (MD) simulations to assess the binding and migration energies of He and defects with each other at various trapping sites such as coherent precipitate interfaces, dislocation jogs and representative grain boundaries. Kinetic Lattice Monte Carlo (KLMC) simulations are used to determine migration mechanisms and diffusion coefficients of substitutional and interstitial helium. KLMC is also used to model helium and vacancy clustering on precipitate interfaces, on dislocation lines and in grain boundaries. The MD and KLMC simulations provide critical information for rate theory and cluster dynamics models that follow point defect and helium transport and partitioning to and recycling between, matrix cavities, precipitates, dislocations and grain boundaries, and the precipitation of helium bubbles and possible conversion to growing voids. The effects of irradiation variables like the irradiation temperature (300-800°C), dpa dose (up to 200 dpa) and dose rate (10⁻⁶ to 10⁻⁷/s) and He/dpa ratio (< 1 to 50) as well as micro structural variations are examined. Prototypical NFA and TMS are both modeled.

Key findings are that a high density of dislocation and NFs traps almost all the helium in fine scale bubbles preventing substantial quantities from reaching and degrading grain boundaries. He partitioning into dislocation and NFs is controlled by their relative sink strengths and helium binding energies. Those model predictions are compared to experimental observations presented in the companion paper.
24th Symposium on Effects of Radiation on Nuclear Materials and the Nuclear Fuel Cycle
"Kinetic Monte Carlo Simulation of Helium Bubble Evolution in ODS Steels"

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Abstract

Oxide Dispersion Strengthened (ODS) ferritic steels are being developed for higher temperature operation for fusion applications. ODS-Eurofer 97 (Fe-9CrWVTa-0.3Y2O3) and ODS-MA957(Fe-14CrTiMo-0.25Y2O3) have greatly improved mechanical properties, such as tensile strength, toughness, fatigue, and creep rupture. Recent neutron irradiation experiments with simultaneous helium implantation indicate that helium transport is also favorably impacted by the nanometer-sized oxide particles, small grain sizes, and high dislocation densities of ODS steels. Describing helium transport in ODS steels requires a 3-dimensional spatially-resolved model, which can take into account the impact of discrete geometric and microstructural ODS features on helium bubble evolution. We have developed a 3-D spatially-resolved helium transport model using the Event Monte Carlo (EKMC) approach. First, a space dependent Kinetic Rate Theory is used to establish helium-vacancy cluster and stable helium bubble nuclei concentrations as a function of implantation depth profile, implantation energies, grain boundaries, dislocation network, and structure temperatures. Using the EKMC model, migration and coalescence of helium bubbles is then simulated subject to interfering oxide particles. Matrix helium bubbles that come into contact with each other are assumed to undergo instantaneous coalescence, which leads to bubble growth based on pressure equilibrium. However, migrating bubbles that are intercepted by oxide particles are trapped and can only grow by newly impinging bubbles, resulting in reduced growth rates of the trapped bubbles compared with matrix bubbles. Helium bubble size- and spatial distributions of the EKMC simulation are compared with recent experimental measurements. To demonstrate the effectiveness of the ODS microstructure on reducing helium bubble growth rates, a comparative EKMC simulation of steels with and without ODS particles is also presented.
In metallic materials low-temperature irradiation produces defect clusters at a relatively high rate, which induces rapid irradiation hardening. Irradiation hardening behavior has been analyzed for dozens of bcc, fcc, and hcp pure metals and alloys irradiated at low temperatures (< -200 °C); focusing on differences and similarities among the irradiation hardening rates $d(<\epsilon_{JS}) / d\epsilon_{PD}$ of different crystalline or material types. Most of ductile metals tested at room temperature showed similar irradiation hardening rates that fall within a narrow band, irrespective of crystal type. Over the whole dose range (0 - 25 dpa) the irradiation hardening rate decreased with dose: the irradiation hardening rates were in the range 100 - 1000 GPa/dpa at the lowest dose of ~0.0001 dpa and a few tens of MPa/dpa or less at about 10 dpa. Log-log plots of the irradiation hardening rate versus dose displayed deflection points in the middle dose range 0.001 - 0.1 dpa, above which the slope of curves decrease. Regression analysis results indicated that in the majority of test materials the exponent of power-law hardening function was about 0.5 for the low-dose regime and about 0.1 for the high-dose regime. Low strength pure metals such as Fe, Cu, and Zr displayed lower hardening exponent values. Two high-temperature metals, pure molybdenum and IN718 alloy, showed nearly zero or negative irradiation hardening rate at low doses (< 0.05 dpa), due to an irradiation softening phenomenon and dissolution of precipitates, respectively. Further, little irradiation hardening is observed in several commercial low alloy steels and model Fe-Cu steels at very low doses (~0.001 dpa), suggesting that a very low dose softening or zero hardening regime can be common for metallic materials.
Interrelationship between True Stress-True Strain and Microstructures of Plastic Deformation Behavior I Neutron-Irradiated or Worm-Rolled Austenitic Stainless Steels

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It is a one of the material problems in aged light water reactors that the loss of ductility due to radiation hardening occurs in the in-reactor components made of austenitic stainless steels. It was believed from many researchers' studies that the radiation hardening was caused by the formation of Frank loops, and the loss of ductility including a decrease of the work hardening coefficient was affected by dislocation channeling. However, authors reported that when the increment of the radiation hardening was converted to the corresponding plastic strain, the true stress-true strain relationship expressed by the Swift's type constitutive equation was similar between irradiated and unirradiated stainless steels. Similar behavior was also reported in Al and IF-steel with sub-urn grain size.

In this study, macroscopic true stress-true strain was compared among the austenitic stainless steels irradiated at about 523 K to dose levels from 1 to 5 dpa and the thermally-sensitized austenitic stainless steel hardened by worm rolling. Both the austenitic stainless steels showed a similar true stress-true strain curve in tensile tests by the conversion mentioned above. Furthermore, when the 0.2% proof stress was higher than about 650MPa, in both irradiated and thermally-sensitized stainless steels the proof stress was about the same to the ultimate tensile stress. Microstructure just before occurrence of plastic instability (necking) was compared between those irradiated and that thermally-sensitized stainless steel using TEM. The effect of microstructure on the occurrence of the plastic instability is discussed.
Influence of Neutron Irradiation on Energy Accumulation and Dissipation during Plastic Flow and Hardening of Metallic Polycrystals

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The irradiation of metallic polycrystals by high energy particles in general leads to hardening and embrittlement. These property changes are frequently studied and reported for stainless steels used for service in nuclear reactors. Not as well studied and reported are some processes accompanying plastic deformation in stainless steels such as the kinetics of accumulation and dissipation of energy, and the thermodynamic peculiarities of martensitic transformations.

In this paper we report results of uniaxial deformation tests conducted using a deformation device where deformation of the sample is carried out directly in a cell of a microcalorimeter. Specimens investigated were 12Crl8NilOTi stainless steel and pure metals (iron, nickel, copper) irradiated to neutron fluences up to $10^{20}$ n/cm$^2$, E>$0.1$MeV at temperatures $<$353K. Additionally the methods of magnetometry, transmission electron microscopy and metallography were used.

As a result of deformation-calorimetric experiments the work of deformation, the dissipated heat and latent (stored) energy versus deformation level were determined. The interrelationship of stress and latent energy were found to depend on the neutron fluence, with latent energy decreasing with increasing neutron fluence.

For metastable steels in the irradiated condition the quantity of martensitic phase (Mf) and its distribution in deformed specimens were investigated. Curves «Mf- s» were constructed and analyzed. The data were used to estimate the thermal effect of martensitic transformation. It was found, that during plastic deformation of pure iron and nickel, the value of dissipated heat can exceed the work of deformation. This unexpected result is explained by channel sweeping of radiation defects as result of their interaction with dislocations. This "abnormal" relationship between work of deformation and latent heat disappears upon annealing of irradiated samples of pure metals, especially iron, in the interval 473-673K.
Hexagonal ducts from the now-decommissioned BN-350 fast reactor in Aktau, Kazakhstan continue to be investigated. As a consequence of the relatively low inlet temperature of BN-350 it has been possible to study radiation-induced evolution at conditions not reached in other studies, allowing investigation at both low and relatively high dpa rates at temperatures below 350°C. It has been shown that void swelling occurs at very low dpa levels (<0.5 dpa) and temperatures as low as 280°C in 12Cr18NiOTi and 08Cr16Ni 1Mo3 steels at low dpa rates characteristic of below-core conditions.. In steel irradiated at higher dpa rates another unanticipated phenomenon has been observed. The macroscopic observation seen during room temperature tensile tests conducted on 12Cr18NiOTi irradiated at 45-55 dpa at 300-350°C is abnormally high plasticity (30% and more), compared to 2-3 % observed at lower values such as 7-12 dpa.

Using a high resolution digital camera focused on the grip section it was observed that the material does not develop a neck but experiences a deformation front that moves along the grip preceded by a plastic deformation wave. Using a combination of digital optical extensometry, magnetometry, metallography and microhardness measurements the peculiarities of this form of plastic deformation have been investigated, showing that a γ→α martensitic transformation is occurring in the deformation area immediately behind the wave front. The high plasticity is a consequence of the martensite-induced hardening pushing the "deformation wave" along the axis of the gauge section. A special method of "magnetic cartography" was developed which allows precise definition of the amount of martensitic phase along the path of the wave. The "true stress - true strain" curves were determined using digital optical extensometry and were analyzed using equations that describe the physics of plasticity.

The results of this study can be used to estimate the plasticity available to highly irradiated fuel assemblies upon extraction from storage and transportation to long term storage facilities.
Irradiation creep of BCC Structural Alloys

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Irradiation-induced dimensional instability is a serious concern in the design and operation of cladding and structural components in advanced nuclear reactor systems. A large amount of data has been obtained on austenitic stainless steels, 316SS and 304SS, and ferritic/martensitic steels, HT-9 and Mod. 9Cr-1Mo, to examine irradiation-induced creep and swelling. It was found that the swelling rate of fee austenitic stainless steels is significantly higher than that of bcc ferritic/martensitic steels. This has led to a perception that crystalline structure plays a significant role in irradiation-induced dimensional changes, and bcc alloys are more resistant to creep and swelling than fee alloys under neutron irradiation. Recently, ample irradiation creep data have become available on bcc vanadium alloys from the Fusion Materials Program, which provide a better picture of irradiation creep in bcc alloys to understand the crystallographic dependence. This paper will analyze and compare the irradiation creep data of two groups of bcc structural alloys, vanadium alloys and ferritic/martensitic steels, and discuss the common characteristics and differences in irradiation creep of bcc metallic materials. Preliminary results have shown that the strong influence of crystalline structure on irradiation-induced swelling and creep observed in fee and bcc iron-based alloys may not be so significant when other factors such as compositional variations, thermo-mechanical treatment, microstructure evolution, and displacement rate, etc are taken in account. This finding is not only important for the theoretical modeling of irradiation creep behavior, but also has significant implications for the engineering and structural design of nuclear reactor components.

The research was sponsored by the Office of Fusion Energy Sciences, the U.S. Department of Energy under contract DE-AC05-00OR22725 with Oak Ridge National Laboratory, managed and operated by UT-Battelle, LLC.
Nano-structured ferritic alloys (NFAs) show great promise for applications in fusion and advanced fission reactors due to their excellent high temperature creep strength and radiation resistance. NFAs are dispersion strengthened by a high density of nm-scale Y-Ti-O enriched nanofeatures (NFs). The objective of this study was to characterize and model the high-temperature static and viscoplastic constitutive laws for as-extruded MA957 and to compare them to data found in the literature for other NFAs and thermal mechanical heat treatment variants of MA957, as well as a ~9Cr tempered martensitic steels (TMS). Flat tensile specimens of as-extruded MA957, with 0.5 x 1.2 x 5 mm gauge sections, were tested in air at a strain rate of 1.33x10^{-6} /s between room temperature to 1000°C. Corresponding creep rates were measured using both strain-rate jump tests and constant load-stress tests from 600°C-1000°C. The strain rate jump (SRJ) tests were carried out by imposing a low strain rate (typically 1x10^{-7}/s) until the stress reached a constant level, followed by a strain rate increase (typically x10) until a new constant stress was achieved. The SRJs were repeated until the specimen failed by rapid tertiary creep. The resulting database, composed of primary creep strains and rates, minimum creep rates and creep rupture times and strains, was used to develop semi-empirical constitutive laws with emphasis on threshold stress models. The threshold stresses were found to be large fractions of the static yield stress (>40%) between 600 and 800°C.
Introduction of Nuclear Standards Development Process in China
Presented by

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Development of a CANDU Fuel Bundle Finite Element Model to Assess Post-Irradiation Mechanical Stress Distributions

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Abstract
A requirement of spent nuclear fuel during interim storage is to maintain its structural integrity at all times to enable its safe and efficient handling during storage, transportation and eventual disposal in a deep geological repository. In Canada, commercial spent fuel is currently stored in licensed wet storage facilities for at least 10 years after irradiation before being transferred to licensed dry storage facilities. The spent fuel may remain in dry storage for up to 100 years before placement in a deep geologic repository. Investigations on the fuel structural integrity evolution during the interim dry storage period are being performed for spent CANDU fuel bundles.

A CANDU nuclear fuel bundle is a cylindrical assembly approximately 0.1 m in diameter and 0.5 m in length containing 28 or 37 fuel elements held together by welding two endplates at both ends. At the weld between the elements and endplates, there is a sharp notch approximately 0.5 mm deep with a notch diameter of less than 10 μm. Significant hydraulic, mechanical, and thermal loads during bundle irradiation in the reactor may lead to bundle deformation which when coupled with the sharp weld notch can result in significant stress enhancement at the notch tip and might activate Delayed Hydride Cracking (DHC) during dry storage.

To better understand the stress levels in CANDU fuel during dry storage, a finite element model of CANDU fuel bundles was developed using ANSYS. The stress distribution in the bundle and the stress intensity factor at each weld notch can be determined for the spent fuel geometry and dry storage conditions. This paper discusses the agreement between the finite element model and three series of validation experiments with unirradiated fuel within the elastic and plastic bundle response regime.
Atomic-Scale Modeling of Dislocation Dynamics in Environment of Radiation Defects

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ABSTRACT
Dynamics of dislocations in realistic environment of existing microstructure defines mechanical properties of crystalline materials. Long-range interactions between a moving dislocation and other defects can be treated within a continuum approach via interaction of their stress and strain fields. However, a vast contribution to mechanical properties depends on the direct interaction between dislocation and other defects and depends very much on the particular atomic-scale structure of the both moving dislocation core and obstacle. In this work we review recent progress in large-scale modeling of dislocation dynamics in metals at atomic level by molecular dynamics and statics. We review the modern techniques used to simulate dynamics of dislocations in different lattice structures, results on temperature, strain rate and obstacle size dependences. Examples are given for bcc, fcc and hcp metals were edge and screw dislocations were interacting with vacancy (loops, voids, stacking fault tetrahedra, etc), self-interstitial clusters and secondary phase precipitates. Attention is paid to interpretation of atomistic results from the point of view of parameterization of continuum models. The latter is vitally necessary for further application in 3-dimensional dislocation dynamics within the multiscale materials modeling approach.

Keywords: dislocation dynamics, radiation damage, localized obstacle, hardening, critical resolved shear stress.
Molecular dynamics simulations have been extensively used in recent years to characterize primary radiation damage formation in the form of atomic displacement cascades. A large database of simulations has been accumulated that describe cascade damage production in single crystal iron using a modified version of the interatomic potential developed by Finnis and Sinclair. This same potential has been used to investigate primary damage formation in nanocrystalline iron in order to have a direct comparison with the single crystal database. A statistically significant number of simulations were carried out at cascade energies of 10 keV and 20 keV and temperatures of 100 and 600K to make this comparison. The variation of cascade size with energy along with the varying grain size demonstrate the influence of grain boundaries as a sink for mobile defects during the cascade cooling phase. Substantially fewer defects survive in the nanograined iron. The influence of the cascade on grain boundary structure and boundary migration was also examined.
Fundamental understanding of the brittle to ductile transition (BDT) and multiscale models of fracture toughness have long been elusive scientific and technical grand challenges. In this work we focus on the characterizing the locally semi-brittle fracture of unalloyed cleavage oriented iron single crystals, as a foundation for understanding and building physical models of the fracture toughness of the complex structural alloys that underpin our technological civilization. We review the first reliable measurements of initiation and arrest fracture toughness in unalloyed iron single crystals oriented for cleavage. The data show that cleavage fracture dynamics are controlled by atomic scale processes that are associated with double kink nucleation on screw dislocations. We show that the semi-brittle cleavage toughness depends on the total yield strength of a material in a way that controls the much higher macroscale fracture toughness of complex structural steels, giving rise to a universal master toughness-temperature curve shape. We propose simple dislocation confinement model to explain this behavior.
Microstructural Characterization of Copper-Containing RPV Materials Irradiated to High Fluences

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Neutron irradiation embrittlement of reactor pressure vessel (RPV) materials is a critical issue for the structural integrity assessment of the RPVs. Especially the embrittlement at high fluences is of great interest for the long term operation of light water reactors because information on the mechanical property changes as well as embrittlement mechanisms is limited at high fluences. In this study, extensive microstructural analyses were conducted on the RPV steels irradiated to high fluences. The steels investigated are base and weld materials fabricated for this project with copper content ranging from medium (0.1wt.%) to high (0.2wt.%). The materials were irradiated in the Halden Reactor with a flux of $5 \times 10^{12}$ n/cm$^2$-s, E>$1$ MeV to fluences up to $1.2 \times 10^{20}$ n/cm$^2$, E>$1$ MeV. After post irradiation mechanical property tests, test specimens for microstructural characterization were machined from the broken Charpy halves, and then three-dimensional local electrode atom probe (LEAP) analyses as well as transmission electron microscope (TEM) observations were performed to characterize the nano-meter scale microstructural changes due to irradiation. LEAP analyses show that, as has been reported in the past, very high number density of solute atom clusters, which consist of copper, nickel, manganese and silicon, were formed in these copper containing materials. Number of copper atoms reach saturation at some fluences, but the numbers of other solute atoms such as Ni, Mn and Si in the clusters keep increasing with fluence. The solute atom clusters are basically so-called copper enriched clusters, but at high fluences, solute atom cluster without copper are also formed. TEM analyses were performed on some materials and formation of dislocation loops were observed though the number density of the loops was relatively low. Linear correlation between the transition temperature shift and the square root of the volume fraction of the solute atom clusters was identified suggesting that the solute atom cluster formation is a primary reason of the embrittlement in this class of copper containing materials. Effects of nickel and phosphorus will also be discussed.

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Very Large Fluence Irradiation Embrittlement of RPV Steel

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Paper shows results from irradiation of specimens from Cr-Mo-V type of steel in surveillance channels of WWER-440 type reactor for more than 15 years. Due to a high lead factor (approx. 13 for base metal and 18 for weld metal), final neutron fluence reached values approx. $12-15 \times 10^{24} \text{m}^{-2} (E > 0.5 \text{ MeV})$, i.e. $7-9 \times 10^{24} \text{m}^{-2} (E > 1 \text{ MeV})$. Notch toughness tests (Charpy V-notch specimens) show that still there is no pronounced saturation of radiation embrittlement and transition temperatures is as high as about 200 or more °C.
Determination of Point Defect Concentration in Neutron-Irradiated Steels by Experimental and Computational Study

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Abstract

A computer simulation and an experimental method were applied to determine a point defect concentration in reactor pressure vessel (RPV) steels irradiated by neutrons. The steel samples had the basic composition of SA508-3 type steels of which the RPV for commercial nuclear power plants are made. The samples were neutron irradiated at 250°C at a fluence of $1.69 \times 10^{19}$ n/cm$^2$ ($E_n > 1.0$ MeV). For estimating the concentration of the point defects in the steels, we used computer simulation methods, including a molecular dynamics (MD) computation and a point defect kinetics model calculation. In order to obtain the primary damage parameters, MD simulations of the displacement cascades in Fe were performed with a 4.7 keV primary knock-on atom (PKA). The energy value, used in the MD simulation, represents the spectral averaged-energy of PKA for a given neutron spectrum. Then, we calculated the point defect concentrations by using the defect kinetics model and the MD calculation results.

In parallel with the computation, we undertook a measurement of the positron annihilation lifetimes (PAL) of the irradiated samples. The PAL test results showed that a certain amount of single vacancies were present in the irradiated steels. By combining the PAL measurement data and the positron trapping model, we calculated the point defect concentration, particularly vacancy one. The assumption in this analysis was that only two types of defect sinks to vacancies exist, which include dislocations and single vacancies. The vacancy concentration derived from the PAL measurement was found to be $7.3 \times 10^{17}$/atom, while that from the computational approach was $1.8 \times 10^{16}$/atom. In spite of some uncertainties about the kinetic and materials parameters, a fair agreement between the two methods was obtained.
The effects of intergranular solutes segregation on embrittlement in terms of ductile-to-brittle transition temperature (DBTT) have been investigated for neutron irradiated reactor pressure vessel (RPV) steels. The four kinds of A533B steels with different bulk P contents ranging from 0.008 wt.% to 0.018 wt.% were neutron irradiated up to a fluence of 1.3 x10^24 n/m^2 (E>1MeV) at 290 °C using a material testing reactor. Published data on several A533B steels were used for a comparison.

Grain-boundary enrichment of P, C, Mo and Ni is detected by a scanning Auger microscopy. The P segregation is facilitated with increasing neutron fluence and bulk content of P. The less C segregation is observed after the irradiation in some cases. The Ni segregation is also increased with neutron fluence. The Mo is almost unchanged by the irradiation.

A correlation between the P segregation, hardening and DBTT shows that neutron irradiation mitigates an embrittling effect of the segregated P, comparing with that in unirradiated and thermally aged steels. The likelihood of intergranular embrittlement in RPV steels at a high fluence is discussed by the present and literature data.

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The paper gives results from the irradiation hardening and embrittlement study of Fe-C-Met type alloys (Fe, Fe-0.2C, Fe-Mn, Fe-0.2C-Mn, Fe-C-1-5 Ni etc.). Study was concentrated on analysis of tensile stress-strain diagrams (Luders deformation, hardening coefficient) and transition temperature shifts. Detailed study was put on microhardness of ferrite and pearlite phases in alloys - different behaviour and trends of both these phases have been found. Some effect of increasing nickel content as well as synergistic effect with chromium and molybdenum has been found. In all types of alloys, existence of carbon that models some steel microstructure has been found as an important factor in the radiation damage value.
Magnox Steel Reactor Pressure Vessel Monitoring Schemes - An Overview

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The steel pressure vessel Magnox reactors were designed in the 1950's and were made from C-Mn plate steel and forgings welded together using a mixture of submerged-arc and manual metal arc weld metals. Each reactor contained surveillance capsules with specimens of plate steel, forgings and the different weld metals to monitor the effects of irradiation on the Charpy impact and tensile properties. Canisters were withdrawn over the operating life and measurements taken. During the lifetime of the fleet, there were developments in testing, observed changes in properties and understanding of the radiation damage process that threatened the operation of the stations. At the time the reactors were designed the concept of fracture toughness was only beginning to be investigated yet during the lifetime of the stations, fracture toughness testing was successfully adopted as standard practice as an input to fracture mechanics based assessment of the vessels. Both hardening and non-hardening embrittlement, the latter due to impurity phosphorus segregation in weld metal, were successfully addressed. At a relatively late stage the contribution of 'thermal neutrons' to embrittlement was identified as being significant and was successfully incorporated into the assessment process. This led to the adoption of sophisticated statistical techniques to assess changes in properties of the most critical construction material - submerged-arc weld metal. A large scale sampling and testing programme of submerged-arc weld metal from a decommissioned reactor validated the assessment process. As a result of successfully addressing these and other challenges when the last two steel pressure vessel stations closed in December 2006, they had achieved life times of nearly 40 years.

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Keywords: steel reactor pressure vessel, monitoring programme.
Final Results from the CARISMA Project on Fracture Mechanical Assessments of Pre-Irradiated RPV Steels used in German PWR

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ABSTRACT:

Pre-irradiated original RPV materials covering all four German PWR construction lines were tested in the CARISMA (Crack Initiation and ARrest of Irradiated Steel Materials) program to create a data base of fracture toughness and arrest values for neutron fluences beyond the EoL range. The new database comprises data from both un-irradiated and irradiated RPV base and weld materials generated by tensile, Charpy-V impact, fracture toughness $K_{JC}$ and crack arrest $K_{IA}$ tests. The test matrix consists of materials with optimized chemical composition and with high Copper or high Nickel content.

Based on the generated and already existing data the RTNDT and the $T_0$ (Master Curve) concepts are applied with specific view on reference temperatures, transition temperature shifts and on possible correlations between the criteria used in both concepts. In this context the consequences of some influencing factors like type and chemical composition of the RPV steel, its manufacturing conditions, and the specimen type and size on the reference temperatures are discussed. Moreover, the test results are assessed with respect to the ASME code and KTA safety standards.

The crack arrest characteristics for these typical RPV materials are also determined in a twofold way by testing Compact Crack Arrest specimens and by evaluation of instrumented Charpy-V impact test data. The available results made a good point that crack arrest is a reliable phenomenon that doubtless exists. It is also shown that the obtained $K_{IA}$ data can be enveloped by applying $K_{IR}$ and $K_{JC}$-curves which are indexed by different reference temperatures.

Finally, the results show that the used RPV materials are well designed in terms of material behavior under irradiated conditions and that optimized manufacture specifications are of great benefit particularly after long operation times.
This paper summarizes results from irradiation embrittlement studies of IAEA Reference Steel JRQ, irradiated in surveillance specimens position in power reactors as well as in experimental reactors within many programmes including IAEA Co-ordinated Research Programmes and some surveillance specimen programmes. Based on these results, trend curves of transition temperature shifts for two major irradiation temperatures - 270 and 288 °C have been constructed. Small scatter of data supports the determination of this JRQ as a material monitor for monitoring and checking irradiation conditions.
Analysis of the Belgian Surveillance Fracture Toughness Database Using Conventional and Advanced Master Curve Approaches

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Abstract
Nuclear reactor pressure vessels are subjected to intense neutron irradiation in the core region, causing embrittlement of base and weld materials. Such embrittlement is quantified by post-irradiation mechanical examinations of specimens contained in surveillance capsules, periodically retrieved from the reactor.

Surveillance specimens are mostly of Charpy-V type and are used within the classical regulatory framework to indirectly estimate the fracture toughness of the irradiated materials. Toughness properties are inferred rather than measured, since the irradiation-induced shift of the fracture toughness curve is defined equal to the shift of the Charpy absorbed energy transition curve at a predefined level (41 J).

This approach entails several weaknesses and does not allow an accurate prediction of the evolution of the actual material properties with neutron irradiation, often leading to over-conservatism and in few cases to non-conservatism.

An advanced surveillance approach, primarily based on direct fracture toughness measurements on surveillance materials in the ductile-to-brittle transition region using the Master Curve procedure, has been applied to several Belgian nuclear power plants in the past 15 years. This has lead to the establishment of a significant fracture toughness database, which consists of 292 data points for 23 material conditions (both unirradiated materials and surveillance capsules). The great majority of test results were obtained on precracked Charpy-V (PCVN) specimens, obtained by reconstitution from previously tested surveillance impact specimens.

In this paper, different temperature normalization approaches are applied to the available data, using (T-T₀), (T-RTNDT) and (T-RT₁₀)- Performing a Master Curve analysis of the database normalized by (T-T₀) shows that data clearly follow the Master Curve formalism. Moreover, using both RTNDT and RT₁₀, it is clear that the static (Kic) and the dynamic (KiA) curves of ASME Section XI both provide an effective lower bound to the measured results, although more conservatism is evident when using RTNDT—Finally, both the conventional (ASTM E1921) and advanced (Multi-Modal) Master Curve analyses of the database clearly demonstrate that normalizing data by (T-RT₁₀) provides the best rationalization of the available information and the most effective representation of the experimental scatter.
The U.S. Nuclear Regulatory Commission (NRC) promulgated Revision 2 of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," in May 1988. That guide details methods that the NRC staff considers acceptable for licensees to use in estimating the effects of radiation on the Charpy V-notch (CVN) impact energy (CVE) of the ferritic steels used in constructing the beltline region of a nuclear reactor pressure vessel (RPV). Radiation damage reduces the ability of these materials to carry load without failure. Consequently, any assessment of the operating safety of the RPV structure should account for the effects of radiation damage.

Since 1988, the nuclear industry has made considerable advances in both the physical understanding of radiation damage processes and the empirical quantification of the effects these processes have on the mechanical properties of RPV steels. Recently, the NRC staff completed an investigation that summarized these advances in the state of knowledge and amalgamated them into a technical basis for an up-to-date version of Regulatory Guide 1.99. The outcome of that study provides the basis of the staff's recommendations on the following matters:

- a formula that can be used to estimate the value of the transition temperature shift at the 30 ft-lb CVE level (AT30) based on the composition of the steel of interest and the conditions under which it has been exposed to neutron irradiation
- a formula that can be used to estimate the value of the upper-shelf energy drop (AUSE) based on the composition of the steel of interest and the conditions under which it has been exposed to neutron irradiation
- the inadvisability of using material- and plant-specific surveillance data to influence or adjust AT30 and AUSE estimates for individual plant assessments
- the margins that should be assigned to the AT30 and AUSE estimates to account for uncertainties
- how the AT30 and AUSE estimates should be adjusted to account for the effects of neutron attenuation through the thick wall of the RPV

In this paper we will provide a summary of the key technical points informing the staff's recommendations for Revision 3 or Regulatory Guide 1.99.
The Microstructure and Hardness Changes of HfIR Irradiated Weld Joint of Vanadium Alloy

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It is recognized that welding procedure of the key technologies for use of V-4Cr-4Ti alloys as a large component. However susceptibility of the alloys to the embrittlement caused by interstitial impurities during welding in highly pronounced. Laser welding technology for the alloys was developed by NIFS (National Institute for Fusion Science) by controlling the flow rate of high purity argon gas. Weld samples of V-4Cr-4Ti alloy (NIFS-HEAT-2) were neutron irradiated in HFIR in direct contact with lithium at temperatures of 450°C and 600°C up to the dose of about 4 dpa. The recovery of irradiation hardening, degradation of impact properties and microstructure of the specimens were investigated after post irradiation and isothermal annealing between 673 and 1073K.

In this study, microstructure and hardness of these samples were compared with the results of neutron irradiation in JOYO.
Looking to the Future of Radiation Damage and Reactor Materials

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