TMS 2021
150th Annual Meeting & Exhibition

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About this Abstract

Meeting 2021 TMS Annual Meeting & Exhibition

Symposium **Mechanical Behavior of Nuclear Reactor Components**

Presentation Title A Novel Displacement Cascade Driven Irradiation Creep Mechanism in Pure Copper

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On-Site Speaker (Planned) Nargisse Khiara

Abstract Scope Metals and alloys used as structural materials in the nuclear core of pressurized water reactors undergo irradiation creep deformation. The mechanical behavior is well characterized at the macroscopic level. Yet, the underlying microscopic mechanisms are still unclear. In situ straining experiments under ion irradiation were conducted on pure copper. We observed that, at high stress levels, dislocations pinned on irradiation induced point defects clusters start to glide once under irradiation. The usual irradiation creep climb induced glide mechanisms were not sufficient to explain the experimental observations. Another mechanism was proposed suggesting that the dislocation glide assisted by irradiation is due to a direct interaction between the displacement cascade and the pinned dislocation. A molecular dynamics study was conducted to test this hypothesis showing that the unpinning induced by a displacement cascade of a dislocation pinned on a loop can be observed under certain conditions of stress, cascade energy and position.

Proceedings Inclusion? Planned:

Keywords Nuclear Materials, Modeling and Simulation, Computational Materials Science & Engineering

OTHER PAPERS PLANNED FOR THIS SYMPOSIUM

A Model for Dislocation Climb and Precipitate Interactions Applied to Creep in Ferritic Steel
A Novel Displacement Cascade Driven Irradiation Creep Mechanism in Pure Copper
Advanced Characterization and Microstructural Analysis of Weldments Performed on Neutron-irradiated Austenitic Steel with H
Atom-probe Study of Nano-hardening Features in Neutron Irradiated RAFM Steels
Atomistic Simulations and Theoretical Modelling of the Yield Behavior of Industrial Tantalum Alloys
Bridging the Length Scales via Femtosecond Laser Machining of Micro-mesoscale Tensile Specimens
Burst Behavior of Accident Tolerant Fuel Cladding Concepts under Simulated Loss-of-coolant Conditions
C-ring Compression of SiC-SiC Cladding at 1200°C with In-situ X-ray Computed Micro-tomography
Challenges to Accurate Evaluation of Bulk Hardness with Nanoindentation Testing at Low Indent Depths
Concurrent Multiscale Modeling of Hydrided Fuel Cladding with Spatially Varied Hydrogen Contents and Irradiation Effects
Controlling Helium Morphology in Pure Metals: Dislocation-helium Interactions
Correlating the Neutron-irradiation Induced Hardening and Solute Nano-clustering in Oxide Dispersion Strengthened Alloys

Creep Behavior of Helium Implanted Submicron Films under Irradiation
Creep Crack Growth Behaviour of Austenitic Stainless Steels Alloy 709 and 316H
Design and Production of a Radiation-resistant Ductile RHEA
Development of Advanced Low N 12Cr Ferritic/Martensitic Steel for Reactor Applications
Development of Modified 3Cr-3WVTa Base Bainitic Steels for Fusion Structural Applications
Dose and Temperature Dependence of Microstructure and Mechanical Properties in Ion-Irradiated PM-HIP Inconel 625
Effect of Cr Concentration On <111> and <100> Dislocation Loop Formation in Fe-Cr Alloys
Effects of Low-temperature Neutron Irradiation and Post-weld Heat Treatment on Tensile Properties of Welded Zircaloy-4
Enabling In-situ Crack Growth Testing and Monitoring in VTR Cartridge Loop Environments
Evaluation of Micromechanical Properties of HT9 Steel under High Energy Heavy Ion Irradiation at Room Temperature
He Ion Irradiation Response of a Gradient T91 Steel
High Temperature Strength of Additively Manufactured Gr91 Steel
High Throughput Assessment of Creep Behavior of Advanced Alloys for Model Development and Validation
In-situ Micro-tensile Studies on the Effects of Ion Irradiation on the Mechanical Properties of Small-grained Alloys
In-situ Observations of the Failure Mechanisms of Hydrided Zircaloy-4 under Different Stress-States
In-situ Scanning Electron Microscopic Observation of Creep and Creep-fatigue of Alloy 709
Irradiation Resistance in Several Multi-principal Element Alloys
Low Temperature Neutron Irradiation and Mechanical Properties of Welded AISI 347
Mechanical Behavior and Radiation Effect in Additively Manufactured 316L Stainless Steel
Mechanical Characterization of Neutron Irradiated HT-9 Heats (ORNL, LANL and EBR II) at LWR and Fast Reactor Relevant T
Mechanical Properties of Additively Manufactured 316L Stainless Steel before and after Neutron Irradiation
Microstructural Effects on the Mechanical Behavior of FeCrAl Alloys
Modeling the Effect of Helium Bubbles, Rigid Inclusions, and Grain Boundaries on Crack Initiation in Nickel
Multiscale Modeling of Creep and Transient Conditions in Steels: Application to HT9 Steel Alloy
Nanomechanical Assessment of a Neutron Irradiated U-10Zr Fuel
Neutron Irradiation Response of SA508 Pressure Vessel Steel Prepared by Powder Metallurgy and Hot Isostatic Pressing
Nitrogen Effects on Radiation Response in HT9 Ferritic/Martensitic Alloys
Novel Small Scale Mechanical Testing Techniques for Nuclear Materials
On the Role of Material Pedigree to Predict Engineering Material Properties
Probing the Mechanical Behavior of Irradiated Materials through Micromechanical Testing
Quantifying Zirconium Embrittlement Due to Hydride Microstructure Using Image Analysis
Simulating the Effects of Nuetron Irradiation on Zirconium Alloys: A Crystal Plasticity Finite Element Approach
Simulation of Intergranular Void Growth under the Combined Effects of Surface Diffusion, Grain Boundary Diffusion, and Bulk
Small Scale Mechanical Testing of Nuclear Fuel and Cladding
Strengthening Elements and Deformation Mechanisms of Ferritic-martensitic Steels
Stress Corrosion Cracking Resistance of FeCrAl Alloys in Light Water Reactor Environments
The Merit of In-situ Environmental TEM for the Study of Tungsten under Fusion-relevant Conditions
The Role of Interface Microstructure and Chemistry on the Bond Strength of Aluminum 6061 HIP-bonded Samples
The Thermo-mechanical Fracture of Chromium-zirconium Systems
Void Swelling and Transmutation in Tungsten Metals and Alloys after Fusion Relevant Neutron Irradiation
Wear Behavior of Incoloy™ 800HT and Inconel™ 617 for High-Temperature Gas-cooled Reactor (HTGR) Applications
α' Precipitation and Hardness Change in Ion Irradiated High Purity FeCr Alloys

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