# Development of Small Punch Testing for Irradiated CuCrZr: Reconstructed Samples

Cholidah Akbar FITRIANI (VTT Technical Research Centre)
Juhani RANTALA (VTT Technical Research Centre)
Asta NURMELA (VTT Technical Research Centre)
Petteri LAPPALAINEN (VTT Technical Research Centre)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Small Punch Tensile Testing (SPT) has gained considerable interest in nuclear materials research, offering the capability to characterize materials with limited sample volumes. Utilizing specimens that are less than 1% the volume of those specified in standard tensile testing (ASTM E8M), SPT provides an efficient and valuable method for preliminary screening and evaluating the properties of irradiated materials. These aspects are important when material availability is restricted, as frequently encountered in fusion materials research. The use of small specimens further allows for the reutilization of samples previously subjected to other mechanical assessments, such as fracture toughness tests. Within the EUROfusion framework, VTT has conducted fracture toughness testing on CuCrZr alloys under both unirradiated and irradiated conditions. Building upon this work and recognizing the advantages of SPT, VTT is currently developing a dedicated small punch tensile testing apparatus at its hot laboratory. In the present phase, this set-up was employed to assess the feasibility of reconstructing and utilizing samples from previously tested fracture toughness specimens to determine the tensile properties of CuCrZr. Comparative analysis was performed between SPT samples fabricated from a pristine CuCrZr block and those reconstructed from fracture toughness-tested samples. The comparative study was performed at room temperature. Following the mechanical tests, thorough fractographical characterizations were performed. The tensile data obtained from SPT were then benchmarked against results from conventional tensile testing to evaluate the suitability of the SPT approach for reconstructed specimens. These findings would help in developing the approach for future tests on irradiated CuCrZr SPT samples.

#12589

# Structural and environmental compatibility of 316L/316Ti–EUROFER97 TIG welds for DONES Target Assembly Backplate

Nerea GARCÍA RODRÍGUEZ (NATIONAL FUSION LABORATORY (CIEMAT))
Marcelo ROLDÁN (NATIONAL FUSION LABORATORY (CIEMAT))
Elisabetta CARELLA (NATIONAL FUSION LABORATORY (CIEMAT))
Iraitz ZUBIAGA (AVS-Added Value Solution)
Anotinio STEENACKERS (AVS-Added Value Solution)
David JIMÉNEZ REY (NATIONAL FUSION LABORATORY (CIEMAT))

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

Dissimilar welds between austenitic stainless steels and EUROFER97 are unavoidable in the DEMO Oriented Neutron Source (DONES) Target Assembly (TA), particularly in the Backplate (BP), where 316L/316Ti piping joins EUROFER97 structural components exposed to extreme neutron flux, lithium velocity and thermal gradients that drive BP lifetime. This work assesses thin TIG (GTAW) butt welds in 1G configuration without filler on 2 mm plates, establishing a unified framework for the structural integrity and liquid lithium compatibility of BP relevant joints.

A 316L–316L reference campaign validated the welding configuration and identified the optimal current window balancing penetration, internal defects and heat affected zone (HAZ) extension. Radiographic inspection showed evolution from lack of fusion at low current (32-34 A) to porosity at high current (45-55 A), with 38-42 A providing the best continuity/porosity compromise. Optical microscopy, SEM and EBSD confirmed austenitic solidification with minor delta ferrite and the equiaxed-to-directionally solidified transition, while room temperature tensile tests and Vickers hardness profiling (200 gf) established the mechanical baseline. Dissimilar 316L–EUROFER97 and 316Ti–EUROFER97 BP representative coupons required higher current (≈65 A) for stable fusion due to EUROFER's distinct thermal properties. Metallographic examination and hardness traverses revealed pronounced microstructural/hardness gradients at ferritic–austenitic interfaces, emphasizing robust preparation/etching needs.

The assessment matrix includes radiographic inspection, EBSD phase/texture mapping across interfaces, and tensile testing for joint level strength, ductility and fracture location. Corrosion performance is evaluated through dedicated liquid lithium exposures (ELIOS-CIEMAT) in sealed AISI 316L containers with high purity Li, assembled in inert atmosphere glovebox, pressurized to 5 bar Ar, heated up to 350 °C, and operated under controlled T/P with real time SCADA monitoring. As welded and Li exposed specimens undergo identical microstructural (optical/SEM/EBSD) and mechanical (tensile/hardness) characterization, enabling direct comparison and quantitative appraisal of DONES TA Backplate dissimilar TIG weld integrity under structural and lithium breeder conditions.

#12681

# Performance Evaluation of Alumina Coatings Applied on 316L and CuCrZr Surfaces in Bolted Joint Structures

Jinho BAE (Korea Institute of Fusion Energy)
Jinho BAE (Korea Institute of Fusion Energy (KFE))
An YOUNGHWA (Korea Institute of Fusion Energy (KFE))
Ku DUCKYOUNG (Korea Institute of Fusion Energy (KFE))
Yoo JEONGWON (Korea Institute of Fusion Energy (KFE))
Seo JIWON (Korea Institute of Fusion Energy (KFE))

### Topics

J. Materials Technology

### Communication type

Poster

### Abstract

This study presents a qualification-oriented test procedure for alumina ($Al_2O_3$) insulation coatings deposited by atmospheric plasma spray (APS) on 316L stainless steel and CuCrZr alloy. The work focuses on bolted interfaces relevant to ITER in-vessel components, where electrical insulation is needed to limit current transfer across metallic contact surfaces during disruption-related transients and to provide input data for electromagnetic (EM) analyses. As an application basis, we consider the supports for the Neutron Activation System (NAS) in-vessel transfer line under development by KODA, which are bolted to bosses on the inner surface of the Vacuum Vessel (VV). In this coating qualification, a two-step approach is adopted. First, APS $Al_2O_3$ coatings are applied to flat coupons with five coating thickness. Electrical insulation behavior depends on thickness is evaluated using insulation resistance and dielectric breakdown tests, supported by microstructural characterization including strength of coating. Second, the selected coating conditions are applied to practical geometrically representative block and clamp specimens. The insulation performance is assessed under simulated joint conditions (controlled variations in tightening torque). The resulting electrical data are processed into equivalent interface parameters for EM modelling, providing engineering inputs for the component design.

# Prediction of Activation and waste generation from EU fusion plant concepts

Priti KANTH (United Kingdom Atomic Energy Authority)
Mark GILBERT (United Kingdom Atomic Energy Authority, Culham Campus, Abingdon, OX14 3DB, United Kingdom)
Sandrine ROSANVALLON (CEA, IRFM Cadarache, France; DEMO Central Team, EUROfusion, Garching, Boltzmannstr. 2 85748 Germany)
Joelle ELBEZ-UZAN (CEA, IRFM Cadarache, France; DEMO Central Team, EUROfusion, Garching, Boltzmannstr. 2 85748 Germany)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

The EUROFusion programme is developing the design of the EU Fusion Pilot Plant (FPP) and First-Of-A-Kind (FOAK). The aim of these devices is to demonstrate fusion and test critical fusion components. Both of these devices will operate with DT fusion plasma, thus creating high-energy neutrons. These high-energy DT neutrons would activate the reactor's structural materials, thereby generating radioactive materials. Previous studies have identified the substantial volume of radioactive waste generated by earlier EU-DEMO models, if waste processing or recycling is not considered. These new fusion devices are based on designs and materials that differ from those of previous EU-DEMO models. In this study, we will calculate the activation and waste generated by the new devices and compare these results with previous calculations. The present work will use the FISPACT-II inventory code to calculate the radioactive inventory in a component as a function of time and then categorise it into waste streams according to regulatory criteria. The inventory calculations also provide radiation source terms and radioactive inventory for other studies, like accident scenarios, decommissioning and dismantling, and waste processing and disposal. This study will also investigate the impact of design changes on these source terms. And finally, the study will propose recommendations for radioactive waste mitigation and minimisation.

# An innovative repair technique for microcracked tungsten combining electrodeposition and hot isostatic pressing

Jingjie SHEN (National Institute for Fusion Science)
Sadatsugu TAKAYAMA (National Institute for Fusion Science)
Takeo MUROGA (National Institute for Fusion Science)
Haochen WANG (Kyoto University)
Yutaro NORIKAWA (Kyoto University)
Toshiyuki NOHIRA (Kyoto University)
Hiroyuki NOTO (National Institute for Fusion Science)

## Topics

J. Materials Technology

## Communication type

Poster

## Abstract

Plasma-facing material tungsten (PFM-W) in fusion reactors is exposed to high heat loads and high radiation, causing various types of damage on its surface. In a typical engineering process, mechanical damage begins as micro-cracks and eventually progresses to extended cracks that can result in large-scale failure, necessitating replacement and disposal. Repairing the damage at a minor stage for sustained use will lead to a reduction of radioactive tungsten waste in future fusion reactors. Regarding this challenge, the authors got an innovative new concept of repairing microcracks induced by pulsed heat loads in tungsten by a combination of electrodeposition and HIP (hot isostatic press). In particular, this concept to repair open cracks by pre-forming a tungsten electrodeposition layer on the surface before HIP processing is unprecedented.

In this study, neodymium (Nd): Yttrium Aluminum Garnet (YAG: $Y_3Al_5O_{12}$) laser (second harmonic, wavelength :532nm) was used to induce cracks on the tungsten. The ITER-grade tungsten was irradiated at a fluence of 100 J/cm2, 200 pulses, and a frequency of 20 Hz, forming ablation craters of approximately 50 μm in width and 50 μm in depth, including high-density of primary and secondary cracks. The cracked tungsten samples were electrodeposited with tungsten of 16.8~22.8μm thickness using a molten CsF-CsCl-WO3 system with the current efficiency of 56.1~76.1% in conditions of electrodeposition temperature: 650℃, current density: 10 mA cm−2, charge density:180 C cm−2, and then were hot isostatic pressed at 1000–1500 °C with a pressure of 200 MPa. The hot isostatic pressed-tungsten samples were sectioned by electrical discharge machining (EDM) and the cross-section of the ablation craters containing the electrodeposited tungsten was evaluated.

In our preliminary experiments for confirmation of repairing microcracks, it was confirmed that the average depth of the microcracks covered with copper was reduced from 7.1 μm to 3.7 μm before and after HIP treatment, indicating the possibility of successful repairing of tungsten using HIP. At the conference, the results of an experiment integrating these technologies to repair microcracks in tungsten by electrodeposition and HIP will be reported. In addition, the potential of the innovative repair technology for fusion divertor maintenance and its broader general engineering applications will be presented.

# Electrodeposited Fe–W monolayers as a basis for graded tungsten enriched coatings for tritium-permeation blankets

Florian LEBENDIG (Karlsruhe Institute of Technology (KIT))
Julia LORENZ-KLEINERT (Karlsruhe Institute of Technology (KIT))
Carsten SCHROER (Karlsruhe Institute of Technology (KIT))
Nils HOLSTEIN (Karlsruhe Institute of Technology (KIT))

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Background. Safe and efficient tritium breeding in water-cooled lithium–lead (WCLL) blankets require robust tritium-permeation barriers (TPBs) that withstand long-term exposure to slowly flowing Pb–16Li. While alumina coatings on reduced-activation ferritic martensitic steel (RAFM) have been Europe's baseline, concerns remain regarding Li uptake, irradiation-induced crystallization, and integrity loss (e.g., anisotropic swelling in $\alpha$ $Al_2O_3$), prompting exploration of non-oxide alternatives.

Objective. We report first steps toward a galvanically produced, graded Fe–W multilayer intended to progressively increase the tungsten content toward high W surface layers for application as a TPB on EUROFER (RAFM) substrates, designed to reconcile thermal mismatch without an electrically insulating interlayer and to enable conformal coating of complex components.

Methods. Iron–tungsten monolayers were electrochemically co-deposited from aqueous sulphate/tungstate/citrate electrolytes on EUROFER. The process space was probed to identify compositional control knobs and to outline a sequential deposition strategy from Fe-rich to W-rich layers, targeting a final, dense W cap.

Results. Monolayers containing 10–40 at.% W were reproducibly obtained. Early results indicate that current density, alongside the electrolyte composition, plays a central role in controlling the tungsten incorporation during co deposition. While the employed electrolytes are operationally safe, their chemical stability requires improvement to ensure reliable processing. A preliminary approach for generating graded layers with progressively increasing tungsten content is outlined. Furthermore, variations in pH, current density, and deposition time were found to influence not only the W content but also the porosity and uniformity of the deposits. Additional factors – including temperature and modified electrolyte formulations or additives – are likewise expected to affect coating quality and are currently being investigated.

Significance. Compared to vacuum plasma-sprayed graded W/EUROFER (millimeter-scale, porous, two-phase), electrodeposition promises micrometer-scale, low-porosity, atom-by-atom growth, better suited to TPB performance and complex geometries. The W top layer is expected to exhibit low solubility and limited Li uptake in Pb–16Li; W oxides are unlikely to be stable in this environment. A known trade-off is electrical conductivity, which does not mitigate magnetohydrodynamic effects; however, the low flow velocity in WCLL may render this acceptable if TPB benefits prevail.

Conclusion. We demonstrate galvanic Fe–W co deposition on EUROFER with tunable composition and outline an initial pathway toward graded Fe–W multilayers as a candidate TPB for the WCLL. Ongoing work focuses on stabilizing and optimizing the electrolyte.

# Native Protium Inventory and Desorption From Stainless Steel Pipework

James PITTARD (UK Atomic Energy Authority, Culham)
Kyle SHIPMAN (UK Atomic Energy Authority, Culham)
Tshepo MAHAFA (UK Atomic Energy Authority, Culham)
Gowri KARAJGIKAR (UK Atomic Energy Authority, Culham)
Zhiheng WU (UK Atomic Energy Authority, Culham)
Iryna BENNETT (UK Atomic Energy Authority, Culham)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Stainless steel pipework is used extensively for transporting gases, ranging from laboratory-scale experiments up to larger scale facilities. Such pipework is known to contain non-negligible amounts of native protium [J. Vac. Sci. Technol. A 21(1) (2002) 167–174], outgassing of which can limit the achievable vacuum and contaminate gas streams. For fusion applications, protium contamination could add additional requirements to an isotope separation system with the potential to change both the concentrations and speciation of isotopologues within gas flows.

To assess the impact of native protium outgassing, an understanding of both the total inventory and the desorption energetics is required. This information can also be used to indicate potential tritium retention and transport through stainless steel components. Therefore, a series of thermal desorption spectroscopy (TDS) measurements have been carried out on samples taken directly from various pieces of SS316 pipework to give a value for inventory and an estimate for the variation between different pieces of pipe. TDS spectra consisted of a single broad peak which gave total inventory values of the order of 10^18 H atoms g (10^1 mol H atoms m^-3). Spectra were fitted with a simple Fickian diffusion model using FESTIM [Int. J. Hydrogen Energy 63 (2024) 786–802] giving an activation energy for diffusion of 0.55 eV which aligns well with expected values. Some samples were measured twice, with the first measurement at a lower temperature meaning a desorption peak in the second measurement could still be seen. The model was able to reproduce desorption spectra for both initial and secondary measurements, whilst its simplicity would aid integration into larger, system scale models.

# Validation of source definitions for neutronics simulations using the JENDL-5 deuteron sublibrary for 5- and 9-MeV beams in the linear IFMIF prototype accelerator (LIPAc)

Kohki KUMAGAI (QST)
Saerom KWON (QST)
Keitaro KONDO (QST)
Kai MASUDA (QST)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

The IFMIF/EVEDA project is being conducted under the Broader Approach agreement between the Government of Japan and EURATOM. As part of these activities, beam commissioning of the Linear IFMIF Prototype Accelerator (LIPAc) is underway in Rokkasho, Japan, to demonstrate the feasibility of high-intensity deuteron beam acceleration in continuous-wave (CW) operation. During the commissioning of the LIPAc radiofrequency quadrupole (RFQ) from 2021 to 2024, deuteron beams with a peak current of approximately 120 mA were accelerated to 5 MeV and delivered at a high-duty cycle of up to 10% to the copper cone of the 1-MW high-power beam dump. The next phase, planned for 2027, will integrate the superconducting RF linac (SRF) to achieve 9-MeV acceleration.

Licensing activities, required under Japanese regulations prior to each beam-operation phase, demand accurate estimation of radiation dose rates to ensure both regulatory compliance and safe operation. A key factor in dose-rate estimation is the definition of radiation sources in Monte Carlo transport simulations. In previous licensing for the 5-MeV operation, source terms were derived directly from experimental measurements. However, these measurements were inherently discrete, and low-energy neutrons could not be measured, introducing unavoidable uncertainties in source characterization.

Recent progress in the development of JENDL-5 has provided a reliably evaluated deuteron sublibrary, enabling more comprehensive modeling of deuteron-induced reactions. Modern transport codes such as MCNP6 and PHITS now support direct use of this sublibrary, allowing fully physics-based source definitions that overcome the limitations of experimental data alone. The JENDL-5 deuteron sublibrary, evaluated using thick-target yield (TTY) experiments with 5- and 9-MeV deuteron beams, is available for copper targets relevant to the LIPAc beam dump.

This presentation reports comparisons of neutron yields obtained from TTY experiments and MCNP calculations using the JENDL-5 deuteron sublibrary. Differences between the experimental and calculated yields are analyzed, and their underlying causes are discussed. Furthermore, the impact of source-term definitions, experimental vs. JENDL-5-based, on dose-rate estimation is assessed. The influence of prompt photon production on dose-rate evaluation is also examined, providing insight into the reliability and applicability of the JENDL-5 deuteron sublibrary for future LIPAc licensing activities. These findings also offer valuable guidance for the design and safety assessment of next-generation deuteron-based accelerator facilities such as DONES and A-FNS.

# Enhanced High-Temperature Strength of RAFM Steel through Tailored Heat Treatment

Chiwon KIM (Korea Institute of Materials Science)
Dan-Woong CHOI (Korea Institute of Materials Science)
Hyung-Kwon PARK (Korea Institute of Materials Science)
Seong-Hoon KIM (Korea Institute of Materials Science)
Hyun-Uk HONG (Changwon national university)
Sang-Woo SONG (Korea Institute of Materials Science (KIMS))
Sung-Jin KWON (Korea Institute of Fusion Energy (KFE))
Chang-Hoon LEE (Korea Institute of Materials Science)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Reduced-activation ferritic/martensitic (RAFM) steels are considered prime structural materials for fusion reactors due to their excellent resistance to neutron irradiation. While their application has primarily focused on blanket structures in ITER, significantly higher neutron doses expected in DEMO reactors require further improvement of their mechanical performance and microstructural stability.

In this study, the heat treatment conditions of conventional RAFM steel were modified to promote fine MX precipitates and refine intergranular Cr23C6 carbides. The normalizing process was adjusted to control the prior austenite grain size, and quenching and partitioning (Q&P) treatment was introduced in combination with optimized tempering to tailor the precipitation behavior.

As a result, the high-temperature tensile strength was improved by 19–28% in the temperature range from room temperature to 550 °C compared with the conventional heat-treated condition. The strengthening mechanisms are discussed from a metallurgical perspective, focusing on grain refinement, precipitation strengthening, and carbide stabilization.

#12893

# Hydrogen-accelerated Creep Degradation at High Temperature in Reduced Activation Ferritic-Martensitic Steel

Hyungkwon PARK (Korea Institute of Materials Science (KIMS))
Yoo JISUNG (Korea Institute of Materials Science (KIMS))
Choi DAN-WOONG (Korea Institute of Materials Science (KIMS))
Kim CHIWON (Korea Institute of Materials Science (KIMS))
Park SEONG-JUN (Korea Institute of Materials Science (KIMS))
Lee TAE-HO (Korea Institute of Materials Science (KIMS))
Suh JIN-YOO (Korea Institute of Science and Technology (KIST))
Lee CHANG-HOON (Korea Institute of Materials Science (KIMS))

### Topics

J. Materials Technology

### Communication type

Poster

### Abstract

Reduced-activation ferritic/martensitic (RAFM) steels are primary structural candidates for fusion blanket systems operating under elevated-temperature hydrogen environments. Although their creep performance under inert atmospheres has been widely studied, hydrogen effects on high-temperature creep deformation remain insufficiently understood. In this study, the creep behavior of a Ta-Ti modified RAFM steel was evaluated at 550 oC under 200 MPa in 10 bar Ar and H2 atmospheres. Hydrogen drastically reduced the rupture life from 5,468 hrs in Ar to 409 hrs in H2, while the total strain to failure and ductile fracture mode remained nearly unchanged. This indicates that hydrogen accelerates creep deformation rather than inducing classical brittle embrittlement. Electron backscatter diffraction (EBSD) analysis revealed a significant reduction in geometrically necessary dislocation density and suppressed formation of low-angle boundaries under hydrogen exposure. Transmission electron microscopy (TEM) observations further demonstrated weakened dislocation-MX precipitate interactions and limited subgrain development. In contrast, carbide stability and cavity formation were not noticeably affected. The results demonstrate that hydrogen enhances dislocation mobility and promotes dynamic recovery, thereby destabilizing the tempered martensitic substructure and shortening the steady-state creep regime. These findings clarify the microstructural origin of hydrogen-accelerated creep degradation in RAFM steels.

# Progress in Material Selection and Assessment for the ITER Glow Discharge Cleaning (GDC) System

Stefano SGOBBA (CERN)
Ignacio AVILES SANTILLANA (CERN)
Berta RUIZ PALENZUELA (CERN)
Roland PICCIN (CERN)
Pavan CHAGANTI (CERN)
Ane BARTUREN MONTES (ITER Organization)
Frantz VILLERS (ITER Organization)
Santiago DOMINGUEZ MEISTER (Tecnalia)
Anastasia DIANOVA PARDO (Tecnalia)

### Topics

J. Materials Technology

### Communication type

Poster

### Abstract

The development and assessment of the Glow Discharge Cleaning (GDC) System, and more specifically of the Permanent Electrodes (PE) mechanical portion is currently ongoing at ITER Organization. GDC involves bombarding the walls with glow plasma in order to reduce and control impurity and hydrogenic fuel out-gassing from plasma-facing components, to provide suitable shielding for the systems behind the GDC electrode during the plasma operation and to perform glow discharge cleaning of the plasma-facing surfaces in support of the boronisation process, which aims at trapping oxygen, carbon, and other contaminants. The GDC electrode consists of two main components: the electrode and the heat sink, interfaced through a ceramic isolation that provides electrical isolation between the different potential components. The GDC electrode should feature the highest possible Thermal Contact Conductance (TCC) between the electrode and the heat sink in order to evacuate the heat from plasma operation. For the heat sink high thermal conductivity is required, achieved through 316LN IG stainless steel (for compatibility with in-vessel material requirements requiring 316-type stainless steels for the wetted surfaces, however featuring very low thermal conductivity), or alternatively CuCr1Zr, coated with high-conductivity pure copper. The electrode material shall feature high thermal conductivity, sufficient electrical resistivity, low thermal expansion, and suitability as a plasma facing material. TZM is being assessed as electrode material for the purposes of this study, although alternative materials such as Inconel 718 are under consideration for the final GDC implementation.

A ceramic material (alumina) under the form of a coating is chosen for the insulation, featuring sufficient electrical insulation resistance, high thermal conductivity, relatively high coefficient of thermal expansion, and radiation resistance. Consequently, flatness, parallelism and surface roughness of the coated electrodes are paramount for a perfect matching with the heat sink. The TCC is dominated by the ceramic isolation which is the largest thermal resistance in the chain.

This paper presents an experimental assessment of the selected material system through a dedicated campaign of TCC measurements conducted under varying contact pressures. In addition, technological challenges associated with the ceramic coating are discussed, with the aim of guiding further optimisation of material selection and manufacturing processes.

# Critical review and rigorous experimental data analysis of hydrogen permeability in Eurofer-97

Urrestizala MARÍA (University of the Basque Country)
Peñalva IGOR (University of the Basque Country)
Alegría NATALIA (University of the Basque Country)
Malo MARTA (CIEMAT - National Fusion Laboratory)
Roldán MARCELO (CIEMAT - National Fusion Laboratory)

## Topics

J. Materials Technology

## Communication type

Poster

## Abstract

Accurate hydrogen isotope permeability data in Eurofer-97 are essential for the reliable design and safety assessment of fusion reactors, where tritium retention and permeation through structural materials play a critical role. However, literature values exhibit a dispersion of up to two orders of magnitude, preventing consensus on bulk transport properties for this RAFM steel.

This work presents a critical reassessment of hydrogen isotope permeability (protium and deuterium) in Eurofer 97 combining a systematic literature review with a rigorously screened experimental dataset obtained using the PermRIG facility at the University of the Basque Country (EHU). Special attention is devoted to identifying the methodological factors responsible for the observed variability, including material characteristics (such as chemical composition and microstructure due to thermal treatment), surface condition, thermal/testing history (such as test sequence and cumulative high-temperature exposure), vacuum quality, background/blank signal control and permeation regime verification.

A strict selection protocol was applied to the experimental data to ensure that only measurements performed under confirmed diffusion-limited regime (DLR) conditions were considered. Regime verification was based on the expected pressure dependence of the steady-state flux (pressure exponent close to 0.5) together with stable, well-defined steady-state signals. Tests exhibiting mixed regimes, unstable fluxes, background signal interference, or surface degradation were excluded. Additionally, XRD analysis before and after permeation tests was used to assess the impact of testing conditions on material structure (e.g lattice strain or in an extreme case, phase transformations).

The results demonstrate that when experimental variables are tightly controlled and only rigorously validated measurements are considered, the apparent dispersion in hydrogen permeability is reduced, revealing a much more consistent behaviour of Eurofer-97 than previously reported. In addition, the study demonstrates that prolonged exposure to hydrogen isotopes at high temperature can induce progressive modifications that significantly reduce permeation flux, helping to explain part of the discrepancies.

Based on the validated dataset, reference permeability values representative of Eurofer-97 in the as-received state (understood as the material in its standard supply condition after conventional normalization and tempering treatments, but prior to any permeation testing, thermal or hydrogen exposure in the experimental facility) are proposed as a conservative upper bound for tritium transport calculations. These findings provide a reliable basis for tritium transport modelling in Test Blanket Modules and highlight the necessity of standardized experimental protocols, to reduce uncertainty in hydrogen isotope transport parameters for fusion applications.

# AI-Assisted Phased-Array Ultrasonic Testing (PAUT) Interpretation for Complex Austenitic Welds in ITER-Relevant Components

Maria ORTIZ DE ZUNIGA (Fusion for Energy)
Andres DANS (Fusion for Energy)
Tito MEGNA (Independent consultant)
Nawal PRINJA (Prinja & Partners)
Ana Maria CAMACHO (National Distance Education University)
Alvaro RODRIGUEZ-PRIETO (National Distance Education University)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Phased Array Ultrasonic Testing (PAUT) is a cornerstone non destructive evaluation (NDE) technique for ensuring the structural integrity of nuclear grade austenitic stainless steel welds used in ITER vacuum vessel (VV). However, PAUT data from thick, anisotropic, austenitic sections remain difficult to interpret due to the effects coming from grain-structure (such as attenuation and beam deflection) and mode conversions. To address these challenges, we present an artificial intelligence (AI) assisted interpretation pipeline developed on historical PAUT A scan datasets from ITER VV relevant manufacturing. The approach leverages artificial neural networks to classify beam responses and automatically identify candidate defect signatures.

The model analyses signal evolution and volume of interest to discriminate conforming from non conforming beams, achieving 99% accuracy on previously unseen A scans. It further localizes defect regions within A scans, enabling inspectors to focus on high risk segments rather than manually analysing entire datasets. This work builds on prior applications of AI to PAUT and NDE workflows in fusion manufacturing, which have shown strong potential for improving defect detection reliability and shortening analysis cycles.

Results demonstrate that integrating AI with PAUT offers quantifiable benefits for quality assurance: reduced manual interpretation time, improved consistency across inspectors, and enhanced traceability of classification decisions. The proposed pipeline is designed to operate as a decision support tool rather than an automated gate, maintaining human in the loop verification essential for nuclear safety class components. We discuss validation needs, qualification pathways, and opportunities for broader industrial data enrichment to increase robustness.

This contribution illustrates a practical route to deploying machine learning enhanced NDE methods within ITER class fabrication environments, supporting safer, faster, and more reliable production of fusion relevant welded structures.

# Modifying Tungsten for an Improved Radiation Shielding Capability and Mechanical Property

Shuhei NOGAMI (A.L.M.T. Corp.)
Naoya MATSUTA (A.L.M.T. Corp.)
Kei MIURA (A.L.M.T. Corp.)
Seiji NAKABAYASHI (A.L.M.T. Corp.)
Makoto WATANABE (A.L.M.T. Corp.)
Shigekazu YAMAZAKI (A.L.M.T. Corp.)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Long-term reliability and cost-effectiveness are critical requirements for fusion reactors beyond ITER. Compact fusion systems have the potential to offer economic advantages, provided that their components and materials can endure higher heat fluxes and neutron irradiation over extended operational durations. In such compact systems, radiation shielding materials must achieve efficient neutron moderation and absorption within limited spatial constraints to ensure adequate protection of superconducting magnets.

Accordingly, it is imperative to develop materials capable of shielding both neutrons across a broad energy spectrum and gamma radiation. Among potential candidates, tungsten-boron systems are promising due to tungsten's high neutron scattering cross-section and gamma ray shielding performance, combined with boron's high thermal neutron absorption cross-section. Nevertheless, conventional tungsten-boron compounds may present challenges in terms of mechanical property and workability, as they typically exhibit low fracture toughness.

To address these issues, A.L.M.T. is advancing the development of novel radiation shielding materials based on the tungsten-boron system. As an initial approach to enhancing both shielding capability and mechanical property, carbon was introduced into the material composition. Candidate chemical compositions were identified through evaluations of shielding capability using the Monte Carlo particle transport simulation code, PHITS. Subsequently, small-scale prototypes were fabricated via powder metallurgy and uniaxial hot pressing, followed by assessments of their fundamental physical and mechanical properties.

Three types of tungsten-boron-carbon compounds were prototyped, and two of these successfully improved both radiation shielding capability and fracture toughness compared to conventional tungsten-boron compounds. This presentation will discuss the influence of carbon addition on both the radiation shielding capability and mechanical properties of the developed materials.

# Corrosion Resistance and Applicability of EUROFER97 in Stagnant Lead-Based Breeding Blanket Environments for Fusion Reactors

Leyla FRANZ (Karlsruhe Institute of Technology (KIT))
Ramil GAISIN (Karlsruhe Institute of Technology (KIT))
Bronislava GORR (Karlsruhe Institute of Technology (KIT))

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Stagnant liquid lead confined in EUROFER97, a reduced-activation ferritic/martensitic (RAFM) steel, is a promising concept for neutron multiplication and shielding in fusion reactor breeding blankets. While EUROFER97 exhibits excellent mechanical and irradiation resistance, its long-term corrosion behavior in contact with liquid lead and under relevant gas atmospheres is not yet sufficiently understood.

This work investigates corrosion mechanisms at the EUROFER97–lead interface, including material dissolution, oxide scale formation, and corrosion kinetics under high-temperature exposure and thermal cycling across the melting point of lead. The combined effects of corrosion and thermomechanical loading due to repeated melting and solidification, as well as thermal expansion mismatch, are assessed by post-exposure microstructural characterization and mechanical testing.

In addition to liquid-metal corrosion, the oxidation behavior of EUROFER97 in helium atmospheres with controlled water vapor content is studied, as unavoidable impurities in purge and cooling gases can strongly influence long-term material degradation. Long-term exposure experiments are conducted in the HEMAT loop at KIT under DEMO-relevant conditions with precisely controlled humidity. Oxide layer growth and composition are analyzed using SEM and EDX at intermediate and final exposure times.

The results provide a systematic assessment of the corrosion resistance of EUROFER97 in both stagnant lead and controlled gas environments and contribute to defining safe operating windows and applicability limits for fusion blanket components.

# Mechanical Interfacing and Vacuum Integrity: Integrating the Target Assembly into the IFMIF-DONES Test Cell

Dénes Zoltán ORAVECZ (HUN-REN EK-CER)
A. ZSÁKAI (HUN-REN EK-CER)
T. DÉZSI (C3D Ltd.)
F.S. NITTI (ENEA Brasimone)
J. MAESTRE (IFMIF-DONES España, Granada )
S. BECERRIL (IFMIF-DONES España, Granada )
J. CASTELLANOS (INAIA, Universidad de Castilla-La Mancha, Toledo)
D. JIMENEZ (CIEMAT, Madrid)

### Topics

J. Materials Technology

### Communication type

Poster

### Abstract

Abstract
The IFMIF-DONES project serves as a critical milestone in the development of fusion energy, specifically designed to qualify candidate materials under high-energy neutron irradiation. The neutron flux is generated in the Target Assembly allocated in the Test Cell. This paper details the system design strategies employed to ensure structural integrity ,while utilizing specialized alloys capable of withstanding intense radiation, mechanical interfaces and vacuum integrity of the Target Assembly allocated in the Test Cell. A primary focus is placed on the internal infrastructure, specifically the updates of the Piping and Cabling Plugs (PCPs) to meet radiation standards and stringent internal safety requirements. Later, we further examined the integration phase, detailing the physical and mechanical interfaces required to securely connect the Target Assembly to the Test Cell. Finally, the study addressed closure and safety protocols, outlining the methodologies for the seamless transmission and exchangeability of diagnostic signals and media from the internal environment to the external through the Test Cell. These advancements collectively aim to provide a robust, secure, and highly functional environment for long-term fusion material testing.

* Corresponding author. oravecz.denes@ek.hun-ren.hu

This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 — EUROfusion). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.

# Scaling-up industrial production of SMART Materials via Mechanical Alloying and Field-Assisted Sintering for First Wall

Jie CHEN (Forschungszentrum Jülich GmbH, Institute of Fusion Energy and Nuclear Waste Management, Plasma Physics (IFN-1), D-52425 Jülich, Germany.)

Andrey LITNOVSKY (Forschungszentrum Jülich GmbH, Institute of Fusion Energy and Nuclear Waste Management, Plasma Physics (IFN-1), D-52425 Jülich, Germany.)

Yuanbin DENG (RWTH Aachen University, Institute for Materials Applications in Mechanical Engineering, Augustinerbach 4, 52062, Aachen, Germany. Institute of Applied Powder Metallurgy and Ceramics (IAPK) at RWTH Aachen e. V., Augustinerbach 4, 52062, Aachen, German)

Uwe FUNK (Dr. Fritsch Sondermaschinen GmbH, Dieselstr. 8, 70736 Fellbach, Germany.)

Ute WILKINSON (Dr. Fritsch Sondermaschinen GmbH, Dieselstr. 8, 70736 Fellbach, Germany.)

Katharina ZOZ (Zoz Group, Maltoz®-Strasse, 57482 Wenden, Germany. )

Hans-Ulrich BENZ (Zoz Group, Maltoz®-Strasse, 57482 Wenden, Germany. )

Giulian KATZENSTEIN (Forschungszentrum Jülich GmbH, Institute of Fusion Energy and Nuclear Waste Management, Plasma Physics (IFN-1), D-52425 Jülich, Germany.)

Mohammad KIANI (Forschungszentrum Jülich GmbH, Institute of Fusion Energy and Nuclear Waste Management, Plasma Physics (IFN-1), D-52425 Jülich, Germany.)

Martin BRAM (Forschungszentrum Jülich GmbH, Institute of Energy Materials and Device, Material Synthesis and Processing (IMD-2), D-52425 Jülich, Germany.)

Jan Willem COENEN (Forschungszentrum Jülich GmbH, Institute of Fusion Energy and Nuclear Waste Management, Plasma Physics (IFN-1), D-52425 Jülich, Germany.)

Christian LINSMEIER (Forschungszentrum Jülich GmbH, Institute of Fusion Energy and Nuclear Waste Management, Plasma Physics (IFN-1), D-52425 Jülich, Germany.)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

SMART (Self-passivating Metal Alloys with Reduced Thermo-oxidation) is being developed as a first-wall material candidate for fusion reactors. Laboratory-scale SMART alloys with composition W–11.4Cr–0.6Y (wt.%) have demonstrated excellent high-temperature oxidation resistance, which is particularly advantageous in accident scenarios such as loss-of-coolant events accompanied by air ingress into the reactor vacuum vessel. Under such conditions, the formation and release of volatile neutron-activated tungsten oxides can be significantly suppressed. Given its proven potential, the investigation of SMART production at an industrial scale is required and falls within the scope of the “a-SMART-FIT” project.

This work reports the scaling up of SMART fabrication using industrial powder-metallurgical routes. Up to date, mechanical alloying of 4 kg SMART powder batches could be successfully achieved, yielding a lattice parameter of ~3.11 Å, comparable to that in laboratory-made SMART powders. An inevitable but reproducible zirconium uptake (~0.4 wt.%) from milling media was detected. The alloyed powders were subsequently consolidated by field-assisted sintering technology. To achieve a satisfying density, ingots were sintered at the temperature of 1550 °C, with a holding time of 10 minutes, and an applied pressure of 55 MPa - conditions more demanding than those employed at laboratory scale. As a result, square-shaped ingots with dimensions of 10 × 10 × 0.5 cm and hexagon-shaped ingots with corner-to-corner distance of 10 cm have been produced successfully, with relative densities exceeding 97%. Microstructural changes were identified in the ingots, characterized by decreasing grain size and homogeneity from center toward corners. The simulation of temperature profiles during sintering, performed with COMSOL Multiphysics, revealed a maximum temperature difference of 97 °C across square ingots after 10 minutes’ holding at 1550 °C. At the same time, a reduced temperature difference of 42 °C was simulated in the hexagon ingots, with the selection of appropriate graphite tooling and geometry. This inhomogeneous temperature distribution is considered as a major contributor to the microstructural gradients observed. Thermogravimetric analyses at 1000 °C in humid air were conducted on specimens extracted from different locations of the ingots to evaluate the impact of microstructural variations on oxidation behavior. As a next step of qualification, plasma exposure experiments

using the PSI-2 facility are also envisaged to further assess the performance of industrial SMART materials under fusion-relevant conditions.

#13086

# Copper-Sealed Microwave Windows for Improved UHV Leak Tightness

Ralph LAUBE (IPP Greifswald)

## Topics

J. Materials Technology

## Communication type

Poster


## Abstract

Reliable vacuum integrity of microwave windows is critical for the operation of ultra-high vacuum (UHV) systems at Wendelstein 7-X (W7-X). In existing assemblies, Viton elastomer seals are frequently used due to their ease of handling. However, leak testing has revealed a time-delayed increase in helium leak rates associated with elastomer-sealed microwave windows. This behavior complicates integral leak detection at W7-X, where a large number of microwave windows are installed at various locations.

This work investigates the helium leak performance of elastomer-sealed microwave window assemblies and presents the development of an alternative sealing concept based on metallic copper gaskets. Copper gaskets provide reliable UHV sealing through plastic deformation between knife edges. In the proposed design, a mica plate is bonded to a specially designed copper gasket using a thin resin layer, maintaining microwave transmission performance while significantly improving vacuum tightness and mechanical robustness. Prototype assemblies were manufactured using standard UHV-compatible materials and tested for vacuum tightness under repeated thermal loading cycles.

Leak testing of the copper-sealed window assemblies demonstrates a substantial reduction in helium leak rates, meeting typical UHV requirements and showing stable performance over multiple cycles. These results confirm that replacing elastomer-sealed microwave windows with copper-gasket-sealed designs offers a viable pathway toward improved vacuum integrity and operational reliability.

# Mechanical Properties of Structural Metallic Alloys for Nuclear Applications Deduced by Three-Point Bend Testing

David BERMUDEZ (SCK CEN)
Dmitry TERENTYEV (SCK CEN)
Thomas PARDOEN (Université catholique de Louvain)
Francisco MIRANDA (SCK CEN)
Alexander BAKAEV (SCK CEN)
Chang CHIH-CHENG (SCK CEN)
David GAROZ (Universidad Politecnica de Madrid)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

The use of miniaturized specimens is increasingly essential for assessing irradiation effects on structural materials in nuclear applications, enabling optimization of irradiation campaigns while minimizing material consumption and nuclear waste. Tensile specimens are commonly used to determine uniaxial mechanical properties, while simpler and smaller bending specimens are used for flexural strength characterization. This work investigates the correlation between miniaturized uniaxial tensile tests and three-point bending (3PB) experiments to determine whether bending specimens can reliably provide tensile yield properties and flow behavior. Miniaturized tensile and 3PB specimens were tested at room temperature and 300 °C for five common nuclear materials: high chromium ferritic martensitic steel EUROFER97, copper-chromium-zirconium according to ITER specification, nickel-based super alloy Inconel600, austenitic stainless steel 316LN, and Titanium grade 2. Tensile results showed strong dependence on crystal structure, with FCC alloys exhibiting high elongation, BCC EUROFER97 high strength with limited elongation, and HCP Titanium intermediate behavior. Temperature effects were significant, with most alloys showing 15–20% reduction in yield and tensile strength, while Inconel displayed Dynamic Strain Ageing effects and Titanium pronounced thermal softening.

Bending tests captured yield and initial hardening, revealing trends consistent with tensile behavior. A reference flexural stress was defined, adapted from methods used for tungsten, to correlate with 0.2% offset tensile yield stress. Stable flexural/tensile correlations were obtained for EUROFER97 (1.83 ± 0.08), CuCrZr (1.65 ± 0.05), and 316L (1.77 ± 0.11), indicating that geometry and elastic–plastic transition dominate this ratio. In contrast, non-stable correlations were obtained for Inconel 600 (1.99 ± 0.28) and Ti (2.23 ± 0.30), which exhibited temperature-sensitive deviations due to strain-rate effects and anisotropic deformation mechanisms. SEM fractography and EBSD analyses confirmed ductile fracture modes and microstructural features influencing mechanical response.

A finite element model, employing an inverse approach with J2 plasticity and GTN ductile damage, was used to reproduce both tensile and bending experiments, capturing stress–strain distributions, plastic volume evolution, and fracture mechanisms. FEM predictions closely matched experimental correlations for materials that showed stable ratios and provided mechanistic insight into multiaxial stress effects in bending. Limitations for rate-sensitive or strongly anisotropic materials were identified, highlighting the need for advanced constitutive models for these cases.

#13161

# Tungsten carbides as near-plasma tritium diffusion barriers compatible for fusion reactors

Liang GAO (Gauss Fusion GmbH, 85748-Garching, Germany)
Markus HERMANSKI (FZJ, Institute of Fusion energy and Nuclear waste management (IFN-1): Plasma physics, 52428-Jülich)
Jan COENEN (FZJ, Institute of Fusion energy and Nuclear waste management (IFN-1): Plasma physics, 52428-Jülich)
Anne HOUBEN (FZJ, Institute of Fusion energy and Nuclear waste management (IFN-1): Plasma physics, 52428-Jülich)
Jacobo ZEGRI (Gauss Fusion GmbH, 85748-Garching, Germany)
Paul STANIEC (Gauss Fusion GmbH, 85748-Garching, Germany)
Sebastijan BREZINSEK (FZJ, Institute of Fusion energy and Nuclear waste management (IFN-1): Plasma physics, 52428-Jülich)
Christian LINSMEIER (FZJ, Institute of Fusion energy and Nuclear waste management (IFN-1): Plasma physics, 52428-Jülich)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Tungsten carbides as near-plasma tritium diffusion barriers compatible for fusion reactors
L. Gao1,2, M. Hermanski2, J. Coenen1,2, A. Houben2, J. Zegri1, P. Staniec1, S. Brezinsek2, Ch. Linsmeier2
1Gauss Fusion GmbH, Parkring 29, 85748-Garching, Germany
2Forschungszentrum Jülich GmbH, Institute of Fusion energy and Nuclear waste management (IFN-1): Plasma physics, 52428-Jülich, Germany

To ensure a self-sufficient and regulatory acceptable tritium fuel cycle for future fusion reactors, tritium uptake by materials from all different sections in a fusion reactor should be limited via, e.g., tritium diffusion barriers (TDB) materials using metal oxides, nitrides or carbides. However, the harsh fusion environments have left very few choices on reactor-compatible TDB materials, especially for the application in the near-plasma region. By positioning the diffusion barrier directly beneath the plasma-facing tungsten, we have in the present work challenged the conventional rear-surface barrier concept and establishes a viable near-plasma tritium management solution for fusion first-wall systems.
Tungsten carbides (WxC) with demonstrated hydrogen-blocking ability [1] and thermal stability [2] are down-selected as near-plasma TDB material. Furthermore, the intermediate thermomechanical properties of WxC [2] between W and ferritic martensite steels (e.g., G91 or Eurofer steels) ensure their potential to be used as joining functional materials. Tungsten carbides coatings are fabricated via SPS sintering on both tungsten and G91 steel substrates and characterized with scanning electron microscopy (SEM), EDX and XRD for their microstructures and composition. Thermomechanical testing will be performed to address the respective adhesion quality, including high heat-flux testing, tensile/shear testing, etc. Results will be analyzed toward a better engineering design admixing other metal carbides for an optimal adhesion quality with G91 steels. Deuterium (D) plasma exposures will also be performed on both tungsten- and steel-based specimens, followed by quantitative deuterium depth profiling using laser-based detection techniques to assess hydrogen-blocking performance under plasma-relevant conditions. In the next phase, monolithic W–WxC–G91 steel mock-ups will be fabricated to demonstrate thermo-mechanical compatibility, interfacial integrity, and tritium retention mitigation at component scale.
Reference:
[1] P. Wang, W. Jacob. Nuclear Instruments and Methods in Physics Research B 329 (2014) 6–13
[2] J. Poetschke, V. Richter, T. Gestrich, A. Michaelis. Grain growth during sintering of tungsten carbide ceramics, International Journal of Refractory Metals and Hard Materials 43 (2014) 309-316

#13163

# Understanding mechanical properties of V-4Cr-4Ti alloys to Tungsten joints through Hot Isostatic Pressing for Fusion Plasma-Facing Applications

Sam CRUCHLEY (School of Metallurgy and Materials, University of Birmingham)
Adam JACKSON (School of Metallurgy and Materials, University of Birmingham)
Moataz ATTALLAH (School of Aeronautical, Automotive, Chemical and Materials Engineering (AACME), Loughborough University)
Michael PARSONS (The Manufacturing Technology Centre (MTC))
Arunodaya BHATTACHARYA (School of Metallurgy and Materials, University of Birmingham)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Reliable joining of plasma-facing and structural materials remains a key manufacturing challenge for future fusion power plants. Tungsten armour must be securely integrated with metallic first-wall components while preserving mechanical integrity under extreme thermal and irradiation environments. For liquid lithium or FLiBe blanket concepts, vanadium-based alloys, particularly V-4Cr-4Ti (V44), are promising first-wall and blanket materials due to their high-temperature strength, low neutron activation, and favourable irradiation resistance. Realisation of these designs therefore depends on the development of scalable joining technologies capable of producing robust W/V44 interfaces. This work investigates Hot Isostatic Pressing (HIP) as an enabling manufacturing route for tungsten–vanadium integration via Diffusion Bonding (DB). Elevated temperature and isostatic pressure promote interfacial diffusion and microstructural evolution, facilitating metallurgical bonding through diffusion and recrystallisation. Mechanical performance of the resulting joints is evaluated over a wide temperature range (room temperature to >900°C), alongside assessment following thermal ageing under fusion applicable temperatures. The advantages and limitations of diffusion-bonded W/V44 joints are critically assessed, defining key development priorities required to advance this joining technology towards fusion deployment. This research was possible due to funding from The Manufacturing Technology Centre (MTC).

# Investigation of the Root Causes of Leakage in Additively Manufactured Water-Cooled Components for Plasma-Exposed Applications in Wendelstein 7-X

Danilo BEIERSDORF (Max-Planck-Institut für Plasmaphysik)

### Topics

J. Materials Technology

### Communication type

Poster


### Abstract

The metal laser powder bed fusion (L-PBF/M) additive manufacturing process has been successfully qualified for the fabrication of water-cooled components for plasma-exposed applications in the stellarator Wendelstein 7-X [1]. Following successful component manufacturing and plasma-exposed operation, additional components were produced for the upcoming operational phase OP2.4.

During incoming inspection of one component type, leakages were detected for the first time by helium and water leak testing. Such leakages had not previously been observed in the more than 40 components manufactured to date. A repeated manufacturing with modified component orientation and positioning within the build chamber did not lead to any improvement, and the components remained leaky. At present, no clear root cause can be identified. The occurrence of such leaks represents a significant reliability risk for additively manufactured, actively cooled plasma-facing components and necessitates a detailed root cause analysis.

This contribution presents ongoing investigations aimed at identifying the underlying damage mechanisms. The analysis program includes the evaluation of manufacturing parameters, assessment of the fabrication process, metallographic investigations, and the determination of mechanical properties.

Based on the results, robust measures to prevent leakage-critical defects will be derived, thereby supporting the reliable application of additively manufactured components in fusion environments.


[1] https://doi.org/10.1016/j.fusengdes.2023.114018

# Mechanical behaviour and microstructure of an ODS EUROFER steel produced by reactive synthesis

Vanessa DE CASTRO (Universidad Carlos III de Madrid)
Maria A. AUGER (Universidad Carlos III de Madrid)
Teresa LEGUEY (Universidad Carlos III de Madrid)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Oxide dispersion strengthened (ODS) reduced activation steels are considered among the most promising structural materials for fusion reactors due to their superior high temperature strength and radiation damage resistance as compared with their non-ODS counterparts. Mechanical alloying of steel powders with yttrium oxide followed by consolidation has been the conventional route to promote the nanometric oxide dispersion in the steel matrix, but it is increasingly avoided due to contamination, inhomogeneity and scaling issues. Therefore, new routes need to be explored. Reactive synthesis is a manufacturing process capable of forming Y-O nanoparticles in-situ during the consolidation stage of steel powders without the assistance of mechanical alloying. In this work, an ODS steel with nominal composition Fe-0.11C-9.0Cr-1.1W-0.40Mn-0.2V-0.12Ta-0.03N-0.27Y has been produced using this method. Surface treatments of gas atomized powders have been applied to create precursor powders containing Y and O, and subsequent precipitation of the nano-oxides has been achieved by hot isostatic pressing consolidation of the oxidized atomized steel powder. The resulting microstructure and mechanical properties are investigated as a function of different annealing treatments applied (normalization and tempering): general microstructure is analyzed by SEM, grain microstructures are analyzed by EBSD, and mechanical properties are measured by profilometry-based indentation plastometry. The results are compared with those obtained for a mechanically alloyed ODS steel of similar composition, as well as with hardness and tensile tests. These findings help to validate the viability of reactive synthesis as a scalable, cleaner alternative to conventional mechanical alloying. Additionally, the work explores the capabilities of profilometry-based indentation plastometry as a reliable technique for extracting high-temperature bulk mechanical properties from small volumes of nuclear materials. Finally, establishing a clear correlation between processing parameters, microstructure, and mechanical performance provides valuable insight for optimizing next-generation ODS steels.

# Evaluation of neutron irradiated effects on advanced nuclear structural materials

Ziwei LI (SCK CEN)
Dmitry TERENTYEV (SCK CEN)
Stefan HOLMSTRÖM (SCK CEN)
Thomas PARDOEN (Université catholique de Louvain)

## Topics

J. Materials Technology

## Communication type

Poster

## Abstract

The development of fusion and Generation IV fission reactors poses significant challenges on the development of structural materials capable to withstand severe neutron irradiation. To address these challenges, innovative structural materials such as High-Entropy Alloys (HEAs) and Oxide Dispersion Strengthened (ODS) steels have been developed. However, their response to neutron irradiation must be thoroughly evaluated beforehand. In this study, A systematic evaluation has been carried out on selected neutron irradiated nuclear structural materials, including baseline nuclear structural material (Eurofer97) and innovative structural materials (HEA and ODS steel). These materials were exposed to neutron irradiation to 1 and 3 dpa under three different irradiation temperatures (300 °C, 550 °C and 650 °C) in the Belgian Reactor 2 (BR2) nuclear reactor at SCK CEN (Belgian Nuclear Research Centre), followed by hardness measurements. By evaluating the changes in hardness before and after irradiation, a comparison among the discussed materials in terms of their resistance to neutron irradiation is provided. The results offer crucial insights into the effects of irradiation dose and irradiation temperature on the ability to withstand neutron irradiation condition of both baseline and advanced structural alloys.

# Understanding low and intermediate temperature volumetric instabilities due to irradiation in Di-tungsten Penta-boride for Spherical Tokamak Centre Column shields.

Mohammed UMAR (School of Metallurgy and Materials, University of Birmingham, Edgbaston)
Alice APPLEBY (School of Metallurgy and Materials, University of Birmingham, Edgbaston)
Samara LEVINE (Tokamak Energy Ltd)
Jason LONERGAN (Massachusetts Institute of Technology)
Aurelie GENTILS (University of Paris-Saclay, CNRS, IJC Lab, Orsay)
Stephanie JUBLOT-LECLERC (University of Paris-Saclay, CNRS, IJC Lab, Orsay)
Sandeep IRUKUVARGHULA (Tokamak Energy Ltd)
Gurdeep KAMAL (Tokamak Energy Ltd)
Arunodaya BHATTACHARYA (School of Metallurgy and Materials, University of Birmingham, Edgbaston)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Di-tungsten Penta-boride ($W_2B_5$) is a promising novel material for shielding high temperature superconductors in future fusion devices, owing to high density and neutron absorption capability. However, essentially nothing is known regarding the effects of cascade-damage irradiation conditions on W2B5's degradation over the temperature range of shield operations. To address this, we systematically irradiated non-stoichiometric $W_2B_5$, varying in phase composition and density due to evolving fabrication techniques, across a temperature range from –196°C to 600°C using energetic ions ($Ar^{2+}$, $Si^{+}$, and $W^{3+}$ ions) up to 3 displacements per atom. Post-irradiation examination via scanning and transmission electron microscopy, along with diffraction technique, enabled a direct comparison of radiation-induced microstructural changes. These changes include susceptibility to radiation induced amorphisation (RIA) and anisotropic lattice parameter swelling which can cause grain boundary microcracking. This work presents a detailed characterisation of RIA phenomenon, radiation- induced extended defects and their correlation of volumetric instabilities as a function of irradiation temperature in $W_2B_5$. This presentation will present a holistic view of radiation damage scenarios in W2B5 ceramics for centre-column shielding, postulating the lower temperature operational limit on this class of material for fusion shielding applications.

Research sponsored by UK Research & Innovation, Engineering and Physical Sciences Council (EPSRC) Prosperity Partnership programme, Grant UKRI564 for Fusion Reactor Shielding Materials (FuReShMa). PhD R&D co-sponsored by Tokamak Energy and the University of Birmingham. Ion irradiations were performed at MOSAIC ion beam facility, JANNuS-Orsay experimental hall, IJCLab.

# Microstructure-Driven Corrosion Behavior of RAFM Steels in tatic Pb-Li Eutectic

Debashish SUR (Materials Science and Technology Division, Oak Ridge National Laboratory, Oak Ridge, TN, 37830)
Yi-Feng SU (Materials Science and Technology Division, Oak Ridge National Laboratory, Oak Ridge, TN, 37830)
Marie ROMEDENNE (Materials Science and Technology Division, Oak Ridge National Laboratory, Oak Ridge, TN, 37830)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Liquid eutectic lead–lithium (Pb–Li) coolants are one of the leading candidates for tritium breeding and neutron multiplication in fusion blanket systems but their compatibility with structural materials remains a significant challenge. Coated reduced activation ferritic–martensitic (RAFM) steels are considered promising structural materials for fusion applications due to their low neutron activation and favorable thermo-mechanical properties, but they remain vulnerable to corrosion in the event of coating degradation during exposure to
liquid Pb–Li. In this study, the corrosion compatibility of recently developed
castable nanostructured alloy (CNA, USA), F82H (Japan), and EUROFER97
(Europe) was evaluated in static Pb–Li eutectic at fusion-relevant temperatures
ranging from 500 to 700 °C for exposure duration of 1000 h, with the objective of elucidating fundamental degradation mechanisms in the absence of coating and flow-induced effects. Post-exposure characterization of the specimens as well as the liquid metal revealed preferential dissolution of alloying elements, Fe and Cr, as well as the strengthening M23C6 carbide precipitates into the liquid metal. This resulted into a corrosion affected zone having a significant grain coarsening with gradients in composition and precipitate content. These results provide benchmark data for understanding corrosion behavior of RAFM steels under static Pb–Li conditions and establish a foundation for future investigations in flowing and irradiated Pb–Li environments, supporting improved lifetime assessment and materials qualification for Pb–Li-based fusion blanket concepts.

This work is sponsored by the Fusions Materials Program and authored by UT-Battelle, LLC, under contract DE-AC05-00OR22725 for the US Department of Energy (DOE).

#13215

# Degradation of Tungsten Carbide Under Irradiation for Application in Spherical Tokamak Centre Column Shielding

Michael WILLIAMS (School of Metallurgy and Materials, University of Birmingham, Edgbaston)
Alice APPLEBY (School of Metallurgy and Materials, University of Birmingham, Edgbaston)
Aurelie GENTILS (University of Paris-Saclay, CNRS, IJC Lab, Orsay)
Stephanie JUBLOT-LECLERC (University of Paris-Saclay, CNRS, IJC Lab, Orsay)
Samara LEVINE (Tokamak Energy Ltd)
Gurdeep KAMAL (Tokamak Energy Ltd)
Sandeep IRUKUVARGHULA (Tokamak Energy Ltd)
Arunodaya BHATTACHARYA (School of Metallurgy and Materials, University of Birmingham, Edgbaston)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

The centre column of a spherical tokamak has limited space, which makes it challenging to shield the high temperature superconducting magnets from damaging neutrons without spatially efficient shielding materials. Tungsten carbide (WC) has been proposed as a promising fusion neutron shielding material due to its good neutron scattering and moderation, high thermal conductivity, and low coefficient of thermal expansion. Despite this, there is little information in the literature regarding how WC will perform under irradiation. To examine the effect of binder material on the amorphisation susceptibility of cemented tungsten carbides, conventional Co binder WC, reduced activation FeCr binder WC, and binder-free WC were ex-situ irradiated with 6 MeV W3+, 2 MeV Ar2+ and 2 MeV Si+ ion beams. These samples were irradiated at temperatures of -194°C to > 600°C at a dose ranging from 0.01 to 3 dpa. Grazing incident X-Ray diffraction with targeted transmission electron microscopy (TEM) were used to reveal the effect of energetic ion irradiations on temperature-dependent degradation scenarios. Using radiation-induced amorphisation as a critical degradation mode, we reveal the lower temperature limits to be used with these materials. The extent of volumetric swelling, anisotropic lattice parameter swelling and its consequences for shield integrity will be discussed.

This work is supported by the UK Research & Innovation, Engineering and Physical Sciences Council (EPSRC) Prosperity Partnership programme, Grant UKRI564 for Fusion Reactor Shielding Materials (FuReShMa), Tokamak Energy. Ion irradiations are performed at MOSAIC ion beam facility, JANNuS-Orsay experimental hall, IJCLab, France.

# Irradiation effects in tungsten-based ceramics for high temperature superconductor shielding components in future fusion reactors

Alice APPLEBY (School of Metallurgy & Materials, University of Birmingham)
Samara LEVINE (Tokamak Energy, Milton Park, Oxford)
Aurelie GENTILS (Université Paris-Saclay, CNRS/IN2P3, IJCLab, Orsay)
Stephanie JUBLOT-LECLERC (Université Paris-Saclay, CNRS/IN2P3, IJCLab, Orsay)
Jason LONERGAN (Department of Materials Science & Engineering, Missouri University of Science & Technology, Rolla, Missouri)
Michael WILLIAMS (School of Metallurgy & Materials, University of Birmingham)
Mohammed UMAR (School of Metallurgy & Materials, University of Birmingham)
Sandeep IRUKUVARGHULA (Tokamak Energy, Milton Park, Oxford)
Gurdeep KAMAL (Tokamak Energy, Milton Park, Oxford)
Arunodaya BHATTACHARYA (School of Metallurgy & Materials, University of Birmingham)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

High-temperature superconducting (HTS) magnets at the centre-column of fusion devices, are highly sensitive to neutron irradiation and thermal damage. This is particularly critical to spherical tokamaks where in-board space availability is limited, thereby, necessitating highly spatially efficient shielding solutions to protect the HTS magnets from fusion's extreme conditions. Tungsten-based carbide and boride ceramics are currently the most promising candidates for HTS shielding. These ceramics will be subjected to neutron bombardment, and a broad range of temperatures, ultimately degrading them. Notably, these materials exhibit temperature-dependent behaviour with irradiation damage, where lower temperatures may induce amorphisation with significant volumetric swelling. Further, intermediate temperatures above Stage-I recovery temperatures are expected to cause lattice parameter swelling, which can cause severe grain boundary microcracking in non-isotropic crystal structures. These effects will additionally be influenced by the manufacturing processes where changes in residual porosity, grain size, morphology, and secondary phases (i.e. the use of binders) may alter damage mechanisms.

Here, using ion irradiation experiments, with light and heavy ions, we will elucidate the overall irradiation tolerance scenarios in WC and ditungsten pentaboride ($W_2B_5$) ceramics from cryogenic to >600 °C. The influence of binders as sintering aids as well as the influence of manufacturing techniques on shield integrity will be presented.

Research sponsored by UK Research & Innovation, Engineering and Physical Sciences Council (EPSRC) Prosperity Partnership programme, Grant UKRI564 for Fusion Reactor Shielding Materials (FuReShMa). Irradiation experiments were performed at MOSAIC ion beam facility, JANNuS-Orsay experimental hall, IJCLab, France.

#13231

# Radiation effects in reduced activation ferritic-martensitic (RAFM) steel welds for fusion first-wall/blanket applications

Contreras Leon LUCAS (University of Birmingham)

Leandro TANURE (Oxford Sigma, Summertown Pavilion, 18-24 Middle Way, Oxford, OX2 7LG, United Kingdom; Nuclear Futures Institute, Dean Street, Bangor, Gwynedd, LL57 1UT, United Kingdom)

Aurelie GENTILS (University of Paris-Saclay, CNRS, IJC Lab, Orsay)

Carsten BONNEKOH (Institute for Applied Materials, Karlsruhe Institute of Technology (KIT))

Arunodaya BHATTACHARYA (University of Birmingham)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Reduced activation ferritic martensitic (RAFM) steels are amongst the leading candidates for fusion first-wall/blanket (FW/B) structures, especially for water/He-cooled and Pb-Li blanket designs. However, a significant knowledge gap due to a lack of literature data on performance degradation of RAFM welds and heat-affected zones (HAZs) exists for fusion-relevant irradiation environments.

In this work, energetic ion irradiation experiments combined with He implantations will be utilised to compare the response of electron beam and tungsten inert gas welded EUROFER97 against the baseline material for irradiation temperatures between 300 and 550 oC. Additionally, under the same conditions, the impact of different Post Weld Heat Treatments (PWHT) on the microstructural stability and mechanical performance of the HAZs will be studied.

Furthermore, the role of He concentration in High Temperature He Embrittlement and the subsequent inter-granular brittle failure due to grain boundary bubble nucleation will be examined. This is achieved through the characterisation of unirradiated and irradiated pre-bent samples. The results generated in this study aim to improve the understanding of irradiation-induced degradation mechanisms in RAFM steel welds and to better understand the true operating temperature window for FW/B applications in future fusion reactor designs.

This research is sponsored by Oxford Sigma.

#13233

# Investigating the effect of thermal aging treatment on the tensile performance of V-4Cr-4Ti

Kieran MARTIN (School of Metallurgy and Materials, University of Birmingham, Birmingham, B15 2TT)
Karen PACHO (School of Metallurgy and Materials, University of Birmingham, Birmingham, B15 2TT)
Adam JACKSON (School of Metallurgy and Materials, University of Birmingham, Birmingham, B15 2TT)
Alice APPLEBY (School of Metallurgy and Materials, University of Birmingham, Birmingham, B15 2TT)
Mary TAYLOR (School of Metallurgy and Materials, University of Birmingham, Birmingham, B15 2TT)
Arunodaya BHATTACHARYA (School of Metallurgy and Materials, University of Birmingham, Birmingham, B15 2TT)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

Vanadium alloys, specifically V-4Cr-4Ti (V44), are leading candidate materials for use in first wall or breeder blanket applications in future nuclear fusion reactors due to their high operating temperature, compatibility with liquid lithium, and resistance to irradiation-induced swelling. The mechanical behaviour of these alloys is highly dependent on interstitial carbon (C), oxygen (O) and nitrogen (N) impurities in solid solution. V44 alloys have been shown to exhibit dynamic strain aging (DSA), where the interstitial impurities diffuse to dislocations, pinning them. DSA manifests as load drops in the work hardening and Lüders extension regimes of an stress-strain curve. During tensile testing at temperatures between 300-700 C and strain rates between 10^-3/s and 10^-5/s, these alloys also show strain rate sensitivity. Novel aging treatments have been developed in an effort to form strengthening Ti-(CON) precipitates and limit the dynamic strain aging behaviour. The interstitial carbon, oxygen and nitrogen in solid solution will be locked into the Ti-CON precipitates via the aging treatment, thereby negating dynamic strain aging and increasing the operating temperature of V44. An in-depth understanding of how dynamic strain aging and interstitial impurities affect deformation behaviour from room temperature to 900 C is lacking, which is necessary for the prediction of in-service material failure. In this study, we perform high temperature tensile tests in high vacuum to develop an understanding of the effect of dynamic strain aging on failure performance. Fractography is performed across the temperature range and is related to tensile performance. EBSD is carried out to understand microstructural evolution during testing. The effect of aging in limiting dynamic strain aging behaviour is investigated and microstructural comparisons are made between standard condition and the aged alloy.

#13239

# Assessing Fusion Performance Limits of Vanadium Alloy E-Beam Welds for Breeder Blanket Applications.

Shurie MCMAHON (School of Metallurgy & Materials, University of Birmingham)
Andrew SOWDER (Electric Power Research Institute (EPRI), Charlotte, NC, USA)
Carsten BONNEKOH (Karlsruhe Institute of Technology (KIT), Institute for Applied Materials, 76344 Eggenstein-Leopoldshafen, Germany)
Arunodaya BHATTACHARYA (School of Metallurgy & Materials, University of Birmingham)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

V-4Cr-4Ti (V44) is a leading candidate structural material for liquid lithium breeder blankets in future fusion reactors due to its low neutron activation, good irradiation tolerance, and wide operating temperature window (~400 – 700 °C). While the irradiation behaviour of baseline V44 is well established, the response of welded joints – critical to breeder blanket and first wall integrity – remains insufficiently understood.

This work investigates the irradiation response of electron beam welded V44, focusing on weld-controlled performance limits relevant to breeder blanket operations. The effects of welding parameters and post-weld thermal history on irradiation-induced hardening, and deformation behaviour are examined across the anticipated operating temperature range (400–750 °C). In addition, controlled elastic bending is employed to impose a known initial stress state, enabling assessment of high temperature helium embrittlement (HTHE) susceptibility of e-beam welded V44 alloys. The experiments are performed using energetic ion irradiations as a surrogate for neutron bombardment. The study will also reveal the role of irradiation dose, He concentration and temperature on microstructure evolution of e-beam welds – the knowledge of which is scarcely available in V alloys. The overall results aim to inform structural design criteria and define safe operating temperature windows for vanadium alloy welds in liquid lithium breeder blanket systems.

# Microstructure optimisation of additively manufactured V-4Cr-4Ti, for fusion first wall/blanket applications

Stylianos KYPRAIOS (School of Metallurgy and Materials, University of Birmingham, Edgbaston, B15 2SE)
Arunodaya BHATTACHARYA (School of Metallurgy and Materials, University of Birmingham, Edgbaston, B15 2SE)
Alice APPLEBY (School of Metallurgy and Materials, University of Birmingham, Edgbaston, B15 2SE)
Miguel ZAVALLA-ARREDONDO (UK Atomic Energy Authority (UKAEA), Advanced Manufacturing Park, Catcliffe, Rotherham, S60 5FX)
Moataz ATTALLAH (School of Aeronautical, Automotive, Chemical and Materials Engineering (AACME), Loughborough University)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

Vanadium alloys, such as V-4Cr-4Ti (V-44) have been considered as structural material for first wall breeders in fusion reactors, thanks to their high operating temperature and excellent corrosion resistance to liquid lithium. Furthermore, unlike reduced activation ferritic martensitic (RAFM) steels, they are non-ferromagnetic, which can potentially simplify plasma confinement designs. Hot rolling is the primary method used to produce such alloys, which comes with the advantage of improving ductility in the alloy. However, such methods require heavy post processing, such as welding and joining, to achieve the necessary geometry. Additive manufacturing (AM), specifically laser powder bed fusion (LPBF), significantly reduces the need for post processing, by producing complex, near-net shape products, while also reducing waste thanks to the reusability of powder, increasing the yield.
The aim of this study is to establish LPBF as a viable method for developing V-44 with properties comparable to that of the hot-rolled alloy, by optimising the microstructure of the additive manufactured samples. Specifically, LPBF has been employed to create a novel methodology for rapid prototyping of highly dense V-44, through varying heat input parameters such as laser power, scanning speed and hatch space. Hot isostatic pressing (HIP) is then used for post processing, to eliminate the remaining porosity and densify the alloy up to 6.05 g/cm3, which is the theoretical density of V-44. To achieve a microstructure comparable to the hot-rolled material, ageing is used for the formation of Ti(CON) precipitates, which strengthens the alloy through precipitation hardening. Scanning electron microscopy (SEM) and electron backscatter diffraction (EBSD) are then used to observe and determine changes in the microstructure. Microhardness measurements and tensile tests at room temperature, as well as at higher temperatures (>500 °C) are performed to quantify the mechanical properties of LPBF-produced V-44 and compared with that of the hot-rolled material.

# Study on Microstructural Evolution and Strengthening Mechanisms in ODS-Cu Processed with Nano-WO3-Assisted Mechanical Alloying

Feifan HE (Institute for Materials Research, Tohoku University, Japan)
Zimo GAO (Institute for Materials Research, Tohoku University, Japan)
Kaiji YAMAMURA (Institute for Materials Research, Tohoku University, Japan)
Hao YU (Institute for Materials Research, Tohoku University, Japan)
Diancheng GENG (Institute for Materials Research, Tohoku University, Japan)
Minha PARK (Institute for Materials Research, Tohoku University, Japan)
Yasuyuki OGINO (Institute for Materials Research, Tohoku University, Japan)
Sosuke KONDO (Institute for Materials Research, Tohoku University, Japan)
Ryuta KASADA (Institute for Materials Research, Tohoku University, Japan)

### Topics

J. Materials Technology

### Communication type

Poster

### Abstract

Oxide-dispersion-strengthened copper (ODS-Cu) is a promising heat-sink candidate for fusion divertors due to its high strength, thermal stability, and irradiation tolerance. However, ODS-Cu produced by mechanical alloying (MA) often suffers from severe powder coarsening and compositional inhomogeneity, problems that become more critical during large-scale manufacturing. Addressing powder coarsening during ball milling is therefore essential for both property enhancement and industrial deployment. In this work, gas-atomized Cu-0.8 wt.%Y-0.8 wt.%Zr powder was used as the starting material, and nanoscale WO3 was introduced as an inorganic process control agent (PCA) and oxygen supplier to suppress powder coarsening effectively. XRD and XPS analyses of powders milled for different durations (24 h, 48 h and 72 h) were employed to elucidate the decomposition pathway of WO3 and the formation/evolution of Y-Zr-O complex oxides. The results indicate that WO3 decomposes completely into W and O after 24 h of milling. The released oxygen is rapidly captured by Y and Zr in the atomized powder, promoting the formation of Y-Zr-O complex oxide dispersoids within the Cu matrix. The nanoscale WO3 facilitates more uniform dispersion and accelerates its decomposition, enabling synergistic strengthening from both W and Y-Zr-O dispersoids. As a result, the ODS-Cu alloy achieves a hardness of 287 HV. This study demonstrates an effective strategy to overcome MA-induced powder coarsening while simultaneously tailoring coherent dispersoids, pushing the strength of ODS-Cu to a new level and providing a solid basis for scalable production.

# Fabrication of TiO?/TiN Dual-Layer Coatings Using Dielectric Barrier Discharge and Atomic Layer Deposition

KIWAMU MOROIZUMI (Graduate School of Energy Science, Kyoto University)
JURO YAGI (Institute of Advanced Energy, Kyoto University)
TAKAYA ISHI (Graduate School of Energy Science, Kyoto University)

## Topics

J. Materials Technology

## Communication type

Poster

## Abstract

In magnetic confinement fusion reactors, liquid lithium (Li) is considered as a liquid metal blanket material. When highly electrically conductive Li flows under a strong magnetic field, magnetohydrodynamic (MHD) drag increases due to the interaction between induced currents and the magnetic field. Furthermore, the strong reducibility of Li leads to corrosion of oxide-based insulation materials. To simultaneously address these issues, a dual-layer coating structure consisting of an electrically insulating titanium oxide ($TiO_2$) layer and a corrosion-resistant titanium nitride (TiN) layer has been investigated.

In this study, plasma nitridation of $TiO_2$ was carried out under atmospheric pressure in an $N_2/Ar/H_2$ mixed-gas atmosphere at 450 °C by applying dielectric barrier discharge (DBD, 3.6–3.8 kV). GD-OES and XRD analyses revealed that nitrogen penetrated to a depth of approximately 500 nm; however, no distinct crystalline TiN phase was detected, and O and N remained in a solid-solution state. This behavior is attributed to the thermodynamic stability of $TiO_2$ and the low diffusion coefficient of nitrogen.

Possible improvements in the DBD process include increasing the nitrogen partial pressure to enhance the driving force reaction, reducing oxygen-related effects by removing impurities from the atmosphere, and promoting diffusion at elevated process temperatures. Nevertheless, even with such optimization, it may be fundamentally difficult to overcome the intrinsic thermodynamic stability of $TiO_2$.

Therefore, the introduction of atomic layer deposition (ALD) is being considered. In this study, F82H steel, a Japanese reduced activation ferritic/ martensitic steel, is selected as the substrate, and $TiO_2$/TiN dual-layer coatings are planned to be deposited on its surface. The deposition will employ an oxidation cycle (TiCl4 precursor–purge–$H_2O$–purge) and a nitridation cycle (TiCl4 precursor–purge–$NH_3$–purge). Based on self-limiting surface reactions, film thickness and composition will be controlled at the atomic-layer level. The objective is to avoid the O and N solid-solution state observed in DBD-treated samples and to form a well-defined dual-layer structure with sharp interfaces.

After deposition, GD-OES and XRD analyses will be performed to evaluate coating structure and composition. Based on the obtained results, structural and functional differences between the ALD and DBD processes will be clarified. The main findings will be presented at an academic conference.

# A novel probe head for near-SOL material exposure experiments at Wendelstein 7-X

Laura DITTRICH (Max Planck Institute for Plasma Physics)
Chandra Prakash DHARD (Max Planck Institute for Plasma Physics)
Christoph VON SEHREN (Max Planck Institute for Plasma Physics)
Dirk NAUJOKS (Max Planck Institute for Plasma Physics)
Arun PANDEY (Max Planck Institute for Plasma Physics)
Carsten KILLER (Max Planck Institute for Plasma Physics)
Dario CIPCIAR (Max Planck Institute for Plasma Physics)

## Topics

J. Materials Technology

## Communication type

Poster

## Abstract

Wendelstein 7-X (W7-X) is an optimized stellarator for improved neoclassical and fast-ion confinement. W7-X features a chain of magnetic islands at the plasma edge, which are used for the heat and particle exhaust. The multi-purpose manipulator (MPM) [1] allows the insertion of probe heads at the outboard mid-plane for specific scrape-off layer (SOL) and material studies in W7-X.
A novel probe head (IPP-MAT5) for the investigation of material erosion-deposition studies has been developed. The probe head has an adapted front shape to be tangential to the last closed flux surface at the position of immersion and tilted in accordance with the pitch angle. On top of the adapted surface, the probe head has a four-sided shallow pyramid shape resulting in an angle of incidence of 8° for the ion flux. It has the possibility to expose 12 material samples of 1 cm × 1 cm simultaneously. The plasma-facing surface is made of tungsten-coated graphite with inserts of boron nitride. The probe head is equipped with two Langmuir triple probes in opposing toroidal directions and one ball-pen probe [2], to obtain local plasma parameters. Three samples can be electrically isolated using boron nitride inserts with the possibility to apply a bias voltage. At multiple positions, the sample and graphite cap temperatures are monitored by thermocouples. This design allows for optimized near-SOL exposures of material samples.
To optimize exposure conditions, EMC3-Lite simulations [3] are performed for selected probe positions and plasma scenarios, to obtain expected heat loads and impact angles on the material samples and surrounding probe head structures. Thermal analyses of the probe head are envisioned using ANSYS.
The goal is to employ the probe head for the material erosion-deposition and transport studies, with a special focus on tungsten and tungsten alloys. The IPP-MAT5 shall be employed in the operational phases 2.4 and 2.5 starting in September 2026. Near-SOL exposure, combined with measured plasma parameters and numerical simulations, are expected to enhance the understanding of plasma-wall interactions in the W7-X stellarator.

[1] D. Nicolai et al 2017 Fusion Eng. Des. 123 960
[2] D. Cipciar et al 2025 Plasma Phys. Control. Fusion 67 115029
[3] Y. Feng and W7-X-team 2022 Plasma Phys. Control. Fusion 64 125012

# Effect of helium on tritium retention in Fe-Cr binary alloys and RAFM steels

Ashwin PERSAUD (School of Metallurgy and Materials, University of Birmingham, Edgbaston)
Rosemary BROWN (UK Atomic Energy Authority, Tritium Fuel Cycle Division, Abingdon)
Aurelie GENTILS (University of Paris-Saclay, CNRS, IJC Lab, Orsay)
Matthew SHARPE (UK National Ion-Beam Centre, University of Surrey)
Arunodaya BHATTACHARYA (School of Metallurgy and Materials, University of Birmingham, Edgbaston)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

Due to tritium scarcity, future fusion energy systems must be fuel self-sufficient. Additionally, proposed first wall structures will experience unprecedentedly high neutron damage. Whilst inducing defect formation, 14 MeV fusion neutrons also transmute, producing significant amounts of helium (He) within the material, (~10 atomic parts per million (appm) He/dpa). This combination of irradiation damage and He results in nucleation of He-vacancy (V) clusters and He-filled cavities. Research shows hydrogen isotopes stabilise cavities, binding strongly with He-V clusters, resulting in tritium entrapment. This risks self-sufficiency and presents safety concerns.

Here, we present two-part experimental results using ion-implantation and deuterium (D) charging to understand the effect of He-induced cavity formation on tritium entrapment, using D as a surrogate, for a Fe-Cr binary alloy and RAFM steel.
Firstly, samples were ion-implanted at JANNUS-Orsay irradiation facility with He implantation levels of 3500 and 7000 appm, over a wide temperature range, 300 to 500 C. Following irradiations, transmission electron microscopy (TEM) lamellae were created by focussed ion-beam milling (FIB) and the evolution of He / He-V clusters were analysed to reveal the dose, temperature and material system dependencies.

Secondly, samples were ion-implanted at the UK National Ion-Beam Centre (UKNIBC) to the same temperature and He concentrations. These were subsequently gas loaded with D at constant temperature in a pressurised loading facility. State-of-the-art ion beam analysis techniques were utilized to reveal the depth distribution of He and D in the materials. The results will present the correlation of D trapping and release with the number density, and size distribution of He-filled cavities in the microstructure.

# Ion-Laser InterAction Mass Spectrometry (ILIAMS) for the ultra-sensitive detection of long lived radionuclides produced by activation from DT neutrons

Carlos VIVO VILCHES (University of Vienna - Faculty of Physics, Isotope Physics)
Kyra ALTINDAG (University of Vienna - Faculty of Physics, Isotope Physics)
Esad HRNJIC (University of Vienna - Faculty of Physics, Isotope Physics)
Silke MERCHEL (University of Vienna - Faculty of Physics, Isotope Physics)
Martin MARTSCHINI (University of Vienna - Faculty of Physics, Isotope Physics)
Lee W. PACKER (United Kingdom Atomic Energy Authority, Culham Campus)
Callum L. GROVE (United Kingdom Atomic Energy Authority, Culham Campus)
Johannes H. STERBA (Center for Labelling and Isotope Production, TRIGA Center Atominstitut, TU Wien)
Matic DOKL (Department of Environmental and Resource Engineering, Technical University of Denmark)
Jixin QIAO (Department of Environmental and Resource Engineering, Technical University of Denmark)
Andreas DRAGOUN (Division of Nuclear Chemistry, University of Cologne)
Erik STRUB (Division of Nuclear Chemistry, University of Cologne)
Karin HAIN (University of Vienna - Faculty of Physics, Isotope Physics)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Future fusion machines will suffer from long-term neutron activation of structural materials. The prediction of the radionuclide inventories is performed with activation codes, such as FISPACT-II, using existing nuclear reaction cross section libraries. To validate those calculations, foils of different materials were irradiated at the Joint European Torus (JET) during the last Deuterium-Tritium (DT) campaigns, and the calculated activities of short-lived radionuclides were compared with those measured by gamma spectrometry [1]. However, radiometric techniques typically provide inadequate sensitivity for long-lived radionuclides, particularly for small samples exposed to neutron fluences substantially lower than expected in fusion power plant environments. Accelerator mass spectrometry (AMS) has been used for 50 years for the ultra sensitive detection of long-lived radionuclides, C-14 being the most prominent example. At the Vienna Environmental Research Accelerator (VERA), we are developing AMS measurements of Nb-91, Nb-94 and Mo-93, with half-lives of 680 years, 20300 years and 4839 years, respectively. These radionuclides are produced in molybdenum containing materials, e.g. stainless steel SS316, due to the reactions of DT neutrons with the stable isotopes Mo-92, Mo-94 and Mo-95. The production of Nb-94 is also possible on niobium, due to neutron capture by its only stable isotope, Nb-93. The main challenge in measuring Nb-91, Nb-94 and Mo-93 by AMS is the potential interference from their respective stable isobars: Zr-91, Zr-94 and Mo-94, and Nb-93. Conventional isobar suppression techniques used in AMS are based on the different energy loss from isobars, but their capabilities reach their limit for these relatively high atomic numbers. In recent years, though, we have proved the capability of laser photodetachment for the suppression of these isobars at the Ion Laser InterAction Mass Spectrometry (ILIAMS) setup at VERA [2]. In the case of Mo-93, this capability has been confirmed by measuring molybdenum oxide samples produced from a Mo foil exposed to thermal neutrons at the TRIGA Mark II reactor at TU Wien. With our current AMS setup, Nb suppression allows us to measure Mo-93 down to 2 μBq. The calculated Mo-93 specific activity for the materials irradiated at JET containing 1%-2% molybdenum is in the 0.04-0.1 Bq/g range. Therefore, using just 0.050 g for the AMS measurements at VERA would yield a sample with a Mo-93 content three orders of magnitude higher than our limit of detection.

[1] Packer et al., Nuclear Fusion 64 (2024) 106059
[2] Vivo-Vilches et al., NIMB 568 (2025) 165847

#13368

# Cryogenic structural materials for next-generation fusion reactors

Klaus-Peter WEISS (Karlsruhe Institute of Technology (KIT), Institute for Technical Physics)
Zahra ABBASI (Karlsruhe Institute of Technology (KIT), Institute for Technical Physics (ITEP))

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

One of the challenges for the high-field superconducting magnet of the future fusion reactor is the development of cryogenic structural materials which are capable of withstanding high magnetic fields as well as having high strength and toughness. It is nearly unattainable to achieve the desired parameters of yield strength (YS) > 1500 MPa and fracture toughness (KIC) > 130 MPa.m1/2 utilizing frequently employed steels such as 316LN, JK2LB, Incoloy 908, and JJ1. The non-magnetic, corrosion-resistant austenitic stainless steel XM-19 (Nitronic50) offers exceptional strength and toughness at cryogenic temperatures, however there are a few shortcomings that need to be addressed before this material is deemed to be the optimal material for fusion reactor compartment design. In this context, the modified Nitronic 50 (N50H) was developed in China through regulation of chemical composition and optimization of casting, forging, and heat treatment steps throughout the production process. The study's main objective is to characterize the properties of both modified and conventional Nitronic50. Tensile testing, fracture toughness assessment, and fatigue analysis are all aspects of mechanical testing that is carried out at cryogenic temperatures. Additionally, material response is correlated with deformation modes, microstructural characteristics, and precipitates stability via optical and electron microscopy.
This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 - EUROfusion). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.

#13389

# Effect of Welding Process Variables on the Delta Ferrite Fraction in Reduced-Activation Ferritic/Martensitic Steel Weld Metal

Yongjoon KANG (Korea Institute of Materials Science)
Kyoung-Hwan KIM (Korea Institute of Materials Science)
Gitae PARK (Korea Institute of Materials Science)
Chan Kyu KIM (Korea Institute of Materials Science)
Yeong-Ho SON (Korea Institute of Materials Science)
Sangwoo SONG (Korea Institute of Materials Science)
Chiwon KIM (Korea Institute of Materials Science)
Hyungkwon PARK (Korea Institute of Materials Science)
Chang-Hoon LEE (Korea Institute of Materials Science)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

Reduced-activation ferritic/martensitic (RAFM) steels, designed by replacing high-activation elements such as Mo, Ni, and Nb in conventional 9Cr heat-resistant steels with low-activation elements including W, V, and Ta, have been under development as candidate structural materials for fusion reactor components. After normalizing and tempering heat treatments, RAFM steels exhibit a tempered martensitic microstructure, where Cr-rich M23C6 carbides are distributed along lath boundaries and prior austenite grain boundaries, and fine (V, Ta, Ti)-rich MC carbides are dispersed within the matrix, resulting in excellent strength and toughness. For the application of RAFM steels to fusion reactor components such as blankets, welding is unavoidable. However, due to solidification and rapid cooling during welding, the weld metal microstructure can differ significantly from that of the base metal, potentially leading to substantial variations in mechanical properties. One of the most critical microstructural differences is the potential for delta ferrite formation in the weld metal. The presence of delta ferrite in the weld metal is well known to deteriorate mechanical properties, particularly impact toughness; therefore, minimizing its fraction is essential. The formation and fraction of delta ferrite are strongly influenced by weld metal chemistry and welding process variables. In this study, the effects of welding process variables, including preheating and heat input, on the delta ferrite fraction were systematically investigated. In addition, the influence of subsequent passes in multi-pass welding on delta ferrite evolution was examined. The results of this study are expected to provide guidance for optimizing welding process variables to minimize the delta ferrite fraction in RAFM steel weld metal.

#13391

# Welding Characteristics of Reduced-Activation Ferritic/Martensitic Steel

Yongjoon KANG (Korea Institute of Materials Science)
Kyoung-Hwan KIM (Korea Institute of Materials Science)
Gitae PARK (Korea Institute of Materials Science)
Chan Kyu KIM (Korea Institute of Materials Science)
Sangwoo SONG (Korea Institute of Materials Science)
Chiwon KIM (Korea Institute of Materials Science)
Hyungkwon PARK (Korea Institute of Materials Science)
Chang-Hoon LEE (Korea Institute of Materials Science)
Yeong-Ho SON (Korea Institute of Materials Science)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Reduced-activation ferritic/martensitic (RAFM) steels are advanced 9Cr heat-resistant steels in which high-activation elements are replaced with low-activation elements, offering excellent mechanical properties and superior resistance to neutron irradiation. Owing to these advantages, RAFM steels are considered promising structural materials for fusion reactor components such as blankets and divertors. Since welding and joining processes are indispensable for the fabrication of these components, the development of suitable welding materials and the evaluation of weld joint properties are essential. The base metal of RAFM steel consists of a tempered martensitic microstructure, where Cr-rich M23C6 carbides are precipitated along prior austenite grain boundaries and lath boundaries, and fine (V, Ta, Ti)-rich MC carbides are distributed within the matrix, resulting in excellent strength and toughness. However, due to solidification and rapid cooling during welding, the weld metal microstructure can differ significantly from that of the base metal, potentially leading to substantial variations in mechanical properties. In this study, K-RAFM steel, developed by the Korea Institute of Materials Science, was used as the base metal. As a fundamental study for the development of welding materials for K-RAFM steel, autogenous gas tungsten arc welding was performed on K-RAFM steel plates under different heat input conditions. The microstructure and mechanical properties of the weld metal were systematically characterized to analyze their correlation.

#13405

# Corrosion behavior of ODS steel in stagnant liquid PbLi

MOISES OÑORO (Universidad Carlos III de Madrid)
TERESA HERNANDEZ (Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (CIEMAT))
FERNANDO J SANCHEZ (Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (CIEMAT))
VANESSA DE CASTRO (Universidad Carlos III de Madrid)
TERESA LEGUEY (Universidad Carlos III de Madrid)
MARIA A AUGER (Universidad Carlos III de Madrid)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

In the development of future fusion reactors, the breeding blanket is a critical component responsible for capturing neutron energy as heat, shielding the vessel, and breeding tritium fuel. Different breeding blanket designs consider using liquid PbLi eutectic alloy, as it can act as a neutron multiplier, tritium breeder, and coolant medium. However, liquid metal corrosion may represent a challenge for the structural materials in contact with the PbLi coolant. Oxide dispersion strengthened (ODS) alloys are considered structural material candidates due to their enhanced performance at high operating temperatures and at high irradiation doses, but their corrosion resistance requires investigation.

In this work, the corrosion behavior of an oxide dispersion strengthened (ODS) ferritic alloy under fusion-relevant conditions is analyzed. An ODS ferritic alloy, with nominal composition Fe–14Cr–2W–0.4Ti–0.3Y2O3 (wt. %), was manufactured by mechanical alloying and consolidated by hot isostatic pressing at 1373 K under an Ar pressure of 175 MPa for 2 hours. The obtained material was hot cross rolled at 1473 K to a ~70 % thickness reduction, annealed at 1273 K for 2 hours, and air-cooled. The material at this stage serves as a reference. The reference material was thermally aged at 873 K for 2000 h in an Ar atmosphere and air-cooled. The reference and the aged materials were immersed in stagnant liquid PbLi at 823 K for 168 hours (1 week) and for 1344 hours (8 weeks). The corrosion effects induced on the ODS alloy before and after aging were characterized by X-ray diffraction, scanning electron microscopy, X-ray energy dispersive spectroscopy, and secondary-ion mass spectroscopy.

Cr and Fe chemical maps and profiles exhibit the most significant evolution, as they are the alloy components with higher solubility in PbLi. Lithium diffusion was observed in both reference and aged materials. The aged material was more affected by corrosion than the reference one, as some surface detachment and a thicker surface oxide layer formation were observed after the longer exposure time. Even though further corrosion studies are required, these results provide valuable information for understanding the corrosion resistance of ODS steels in liquid PbLi.

# Development of Innovative In-vessel Shielding for Fusion Tokamaks using Borated Water and Porous Tungsten

Chris ELSE (Kyoto Fusioneering UK Ltd)
Paul BARRON (Kyoto Fusioneering Ltd.)
Colin BAUS (Kyoto Fusioneering UK Ltd)
Luigi CANDIDO (Kyoto Fusioneering UK Ltd)
Suneui LEE (Kyoto Fusioneering Ltd.)
John MCGRADY (Kyoto Fusioneering Ltd.)
Italo GODOY-MORISON (Kyoto Fusioneering Ltd.)
Mario OLIVER (Kyoto Fusioneering UK Ltd)
Ben RAEVES (Kyoto Fusioneering UK Ltd)
Alejandro REMIRO (Kyoto Fusioneering UK Ltd)
Carli SMITH (Kyoto Fusioneering America Ltd.)
Jun TAKAMINE (Kyoto Fusioneering Ltd.)
Jack TAYLOR (Kyoto Fusioneering UK Ltd)
Luke TAYLOR-KING (Kyoto Fusioneering UK Ltd)
Andrew WILSON (Kyoto Fusioneering UK Ltd)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Future fusion power plants require in-vessel shielding solutions that provide effective neutron and gamma attenuation, while balancing structural integrity, thermo-mechanical and activation requirements. Kyoto Fusioneering has developed and evaluated a concept based on porous tungsten with borated water as a flowing coolant and shielding contributor and compared this against the reference case of boron carbide, in conditions analogous to UKIFS's STEP machine.

An initial literature review was used to survey and down select candidate shielding approaches. Early neutronics calculations indicated the outsized shielding performance of the tungsten and borated water in comparison to the alternatives, across assessed depths and plasma configurations. Greater shielding effectiveness can reduce machine diameter and ease vessel integration challenges, leading to the selection for further appraisal.

The material was integrated into a behind-blanket shielding design at conceptual level. Neutronics analyses were performed to quantify neutron and gamma attenuation performance. Activation studies evaluated induced radioactivity, decay heat, and waste classification to understand maintenance and decommissioning implications. Structural and thermo mechanical analyses, including seismic loading, were conducted to examine loading under representative fusion plant conditions. In addition, a lifecycle assessment (LCA) was carried out to assess the embedded carbon of the tungsten system, alongside an assessment of cost and manufacturing readiness.

The results indicate that porous tungsten and borated water combined offer a promising solution for fusion facilities, with benefits in packaging and shielding performance. This work points towards an enabling technology for next generation in-vessel shielding systems, and improved machine viability.

Keywords: In-Vessel Shielding; Fusion; Neutronics; Structural Analysis; Life Cycle Assessment

#13445

# The combined microstructure of ODS Eurofer and Eurofer steel for optimization of strength and ductility properties

Ludek STRATIL (Institute of Physics of Materials, Czech Academy of Sciences)
Hynek HADRABA (Institute of Physics of Materials, Czech Academy of Sciences)
Filip SISKA (Institute of Physics of Materials, Czech Academy of Sciences)
Eliska SISKA VIRAGOVA (CEITEC BUT, Brno University of Technology; ThermoFisher Scientific)
Luca BERTOLLA (Institute of Physics of Materials, Czech Academy of Sciences)
Zdenek CHLUP (Institute of Physics of Materials, Czech Academy of Sciences)
Monika VILEMOVA (Institute of Plasma Physics, Czech Academy of Sciences)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

The oxide dispersion strengthened steels are assumed as a structural material of the components of future fusion or fission reactors. The distinct microstructural features of the ODS steels are fine grains and dense dispersion of oxide particles, which ensure their superior tensile and creep strength up to high temperatures over the steels without oxide dispersion. Nevertheless, the ODS steels suffer by low level of toughness which limits their applicability. This work is aimed on the preparation of a composite steel combining microstructures of the ODS Eurofer steel and Eurofer steel. In this microstructural design, the role of the ODS Eurofer steel and Eurofer steel is a matrix which controls strength level of the composite steel and a toughening phase with higher ductility regularly distributed in the matrix, respectively. The composite steels with various fractions of high ductility phase as well as the ODS Eurofer and Eurofer steels were prepared by powder metallurgy route. The microstructures, mechanical and fracture properties of consolidated steels were assessed. The rule of mixtures was determined on the basis of the properties of the ODS Eurofer and Eurofer steels. The increasing fraction of high ductility phase of the composite steels led to the decrease of strength properties and to the increase of deformation properties and resistance to crack propagation. The composite steels achieved considerable improvement of fracture toughness owing to the effect of strain partitioning causing larger extent of process zone in the vicinity of the crack tip.

# Quantifying Orientation-Dependent Interfacial Stress Coupling in W–Cu Heterostructures for Fusion First Wall Materials

Min PAN (Southwest Jiaotong University)
Yang HU (Southwest Jiaotong University)
Zhang QIAN (Southwest Jiaotong University)
Wen SHULONG (Chengdu University of Information Technology)
Deng HUIQIU (Hu nan University)

## Topics

J. Materials Technology

## Communication type

Poster

## Abstract

Tungsten–copper (W–Cu) layered structures are widely considered for fusion first wall components, where interfacial stress redistribution plays a critical role in structural integrity under thermo-mechanical loading. In this work, first-principles calculations are conducted on W(110)/Cu(110), W(110)/Cu(111), and W(100)/Cu(100) interfaces with systematically varied Cu thickness to quantify interface-induced stress evolution. An excess in-plane stress framework is applied to separate intrinsic interfacial stress from lattice-constrained slab contributions, and a layer-resolved stress analysis is employed to evaluate the spatial extent and effective thickness of stress transfer across the interface. The results demonstrate clear orientation-dependent mechanical behavior: the geometrically compatible W(110)/Cu(110) interface exhibits weak and rapidly screened stress perturbation, whereas W(110)/Cu(111) develops pronounced anisotropic stress accumulation that penetrates deeper into the Cu overlayer and evolves nonlinearly with thickness. These results provide quantitative guidance for selecting crystallographic configurations and thickness design in W–Cu first wall structures to improve mechanical stability and service reliability.

# Thermo-Mechanical Validation of Bolted ITER Blanket Cooling Manifold Supports with Alumina-Coated Interfaces

Adrià GALLIFA TERRICABRAS (Fusion for Energy (F4E), Barcelona)
Luciano SICILIA (Fusion for Energy (F4E), Barcelona)
Joaquim CREUS PRATS (Fusion for Energy (F4E), Barcelona, Spain)
Elena FERNANDEZ CANO (Fusion for Energy (F4E), Barcelona)
Laurent GUERRINI (Fusion for Energy (F4E), Barcelona)
Franck DECHELETTE (ITER Organization, St. Paul-lez-Durance)
Stefan NICOLICI (ITER Organization, St. Paul-lez-Durance)
Ryan HUNT (ITER Organization, St. Paul-lez-Durance)
Alessandro BONITO OLIVA (ITER Organization, St. Paul-lez-Durance)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

The ITER Blanket Cooling Manifold (BCM) consists of nearly 10 km of 316L stainless-steel piping with more than 2000 supports, arranged in 363 cooling circuits feeding 440 blanket modules and transferring up to 736 MW of thermal power to the Tokamak Cooling Water System. The system operates under severe thermo-mechanical, electromagnetic, and neutronic loads, with strict zero-leak requirements and tight geometric tolerances. To limit electromagnetic loads, the majority of pipes must be electrically insulated from their supports, making the pipe–support interface a critical mechanical and thermal design element.

The BCM supports must ensure structural integrity and stable thermal contact while maintaining electrical insulation. Alumina coatings are therefore introduced at the interfaces and directly govern both mechanical behaviour (via friction) and thermal contact conductance (TCC) of the pipe–support interface.

A new baseline design based on bolted supports intended to clamp the BCM pipes has been developed. In this concept, alumina-coated interfaces electrically insulate the supports from the pipes while preserving mechanical performance and controlled heat transfer.

An experimental campaign is presented covering both full-scale multi-pipe support mock-ups and simplified single-pipe configurations. Axial mechanical tests (static and cyclic), resulting in shear loading of the alumina–metal interface, are reported together with measurements of TCC via infrared thermography. The influence of tightening torque is investigated over a range of preload levels. In parallel, a characterization of Detonation Gun (D-Gun) alumina coatings is presented, providing insights into coating integrity and mechanical behaviour under representative loading conditions.

The results allow the validation of several manufacturing and design parameters, including geometrical tolerances, effective tightening torque, and resulting friction coefficients. Full validation under operating conditions might require additional tests at high temperature and under vacuum. Beyond ITER, the findings are applicable to other mechanical systems employing high-performance alumina coatings where coupled mechanical and thermal interface behaviour is critical.

# Feasibility Study on Lithium Isotope Separation Using Crown Ether for Tritium Breeder

Young Ah PARK (Korea insitute of fusion energy)
Yi-Hyun PARK (Korea insitute of fusion energy)
Mu-Young AHN (Korea insitute of fusion energy)
Woong-chae KIM (Korea insitute of fusion energy)

### Topics

J. Materials Technology

### Communication type

Poster

### Abstract

Tritium, an essential fuel for fusion energy, is produced through a nuclear reaction between neutron and the lithium contained within tritium breeder. While 6Li reacts across a wide range of neutron energies and contributes positively to energy production via exothermic reactions, 7Li reacts only with fast neutrons and undergoes endothermic reactions, leading to relatively lower energy efficiency. Therefore, a tritium breeder with a high concentration of 6Li is essential to enhance tritium breeding efficiency, as it enables the effective utilization of neutrons across a wide range of neutron energies and maximizes thermal energy output through its favorable exothermic reactions. However, since the natural abundance of 6Li is only about 7.5 %, using natural lithium directly poses limitations in tritium breeding efficiency. Consequently, the development of isotope separation technologies to enrich 6Li from natural lithium is indispensable. This study aims to evaluate the feasibility of lithium isotope separation using Crown Ethers as a selective extraction medium.

This study attempted to derive maximize lithium isotope separation efficiency based on physical separation method, focusing on separation conditions and the specific structure of the Crown Ether. Crown Ethers consists of a repeating unit having –O–CH2–CH2– structure, where multiple oxygen atoms act as multidentate ligands for alkali metal cations. The lone pairs of electrons on each oxygen atom form coordination bonds with alkali metal cation, creating stable complexes where the ion is encapsulated within the center of the ring. The cavity diameter of a crown ether varies depending on the number of oxygen atoms, which determines the specific types of alkali metal cations that can be stably coordinated based on their ionic radii. The concentration of each isotope and separation factor ($\alpha$) were quantitatively analyzed using Inductively Coupled Plasma Mass Spectroscopy (ICP-MS).

In this study, the experimental design and the results of the lithium isotope separation performance will be addressed.

# Oxide nano-dispersion in ODS cold sprayed steels for advanced fusion reactors

Vanessa DE CASTRO (Universidad Carlos III de Madrid)
Teresa LEGUEY (Universidad Carlos III de Madrid)
Emma KRUCK (Université Paris-Saclay, CEA, Service de Recherche en Matériaux et procédés Avancés)
Jean HENRY (Université Paris-Saclay, CEA, Service de Recherche en Matériaux et procédés Avancés)
Yann DE CARLAN (Université Paris-Saclay, CEA, Service de Recherche en Matériaux et procédés Avancés)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

Nanostructured reduced activation oxide dispersion strengthened (ODS) ferritic steels are promising candidates for structural components of advanced fusion reactors. These steels exhibit enhanced creep and mechanical resistance, as well as thermal stability, allowing an increase on their operating temperature as compared with their non-reinforced counterparts. Moreover, stable Y-rich oxide nanoparticles homogeneously dispersed in the steel can be efficient trapping sites for irradiation induced transmutation products, in particular He and bubbles formed thereof, as well as point defects, improving radiation resistance. ODS steels are conventionally produced by mechanical alloying (MA) followed by hot isostatic pressing (HIP) consolidation. However, alternative production routes are also being explored. In this work, the microstructures of ODS steels produced using either cold spray additive manufacturing (CSAM) or MA+HIP, have been characterised focussing on the nanoparticle dispersion. The main objectives are to determine the nature, size and spatial distribution of nanoparticles and other secondary phases present to aid in establishing the best processing route.

Four ODS ferritic steels with the same composition, Fe-13.1Cr-0.12Ti-0.082Mn-1W-0.077Y-0.22O (wt%), were analysed. Two of them were produced by CSAM and subsequently either thermally treated (CSAM+TT), or HIPed (CSAM+HIP). These steels were compared with two other steels with similar nominal composition but produced following the more traditional MA+HIP route. The difference between them was the size of the starting powder; non-sieved or sieved to sizes < 50 μm.

TEM samples of these ODS steels, prepared either by FIB or electropolishing methods, were analysed using (S)TEM-EDS and HREM techniques. The microstructure of the alloys was compared to determine the most promising material in terms of the oxide nano-dispersion obtained. A homogeneous dispersion of ternary Y, Ti nano-oxides is present in all investigated alloys. Smaller nanoparticle sizes and higher number densities are found for the CSAM+TT ODS steel, being the average density of 3 x 10^23 m-3, one order of magnitude higher than the one determined for the CSAM+HIP ODS steel.

#13569

# Functional materials for fusion power plants, current status of the European program

Rafael VILA (CIEMAT, Fusion Technology Division)
Rafael VILA (CIEMAT, Fusion Technology Division)
León EDGAR (CIEMAT, Fusion Technology Division)
Group EUROFUSION MAT (EUROfusion Consortium, Garching, Boltzmannstr. 2, D-85748, Germany)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Functional materials such as fused silica, sapphire, and diamond are essential in fusion reactors, serving as optical windows, dielectric insulators, and key components for diagnostics, heating, and current drive (H&CD) systems. Within the EUROfusion Work Package Materials - Functional Materials (WP-MAT-FM), coordinated research has focused on assessing neutron-induced damage in these materials to ensure their viability in high-radiation environments, like DEMO

Over the past five years (2021–2025), substantial progress has been achieved in characterizing neutron/gamma irradiation effects. Irradiation campaigns in fission reactors and gamma sources have exposed samples to fluences up to 10^21 n/cm$^2$ and high gamma rates, recreating fusion-relevant conditions.

In fused silica, neutron irradiation generates point defects and color centers, increasing optical absorption. It has been found that these effects primarily degrade transmission in the UV, while permitting its use as a visible-range window. Notably, damage saturation occurs around 10^20 n/cm$^2$. A comprehensive comparison of data across various doses and irradiation temperatures is presented.

Sapphire ($Al_2O_3$) and alumina exhibit increased optical absorption and dielectric loss tangent, adversely affecting their performance as optical systems or RF windows for ion cyclotron resonance heating.

Significant effort has been devoted to develop nano-grained spinel (MgAl2O4) but, as presented, ion irradiations did not show improvement on radiation resistance.

Diamond is valued for its excellent thermal conductivity and low loss at microwave frequencies; however, it exhibits a low radiation resilience, and suffers rapidly from graphitization and vacancy-interstitial complexes at low neutron doses, with strong Raman peak intensity reduction. Development of single-crystal diamond has started in EUROfusion to try to overcome the limits found in polycrystalline CVD diamond.

Future challenges include achieving tolerance to DEMO-relevant fluences (higher dpa), mitigating synergistic effects of neutrons, heat and electromagnetic fields, and developing mitigation approaches like doping or optimized operation temperature. Until IFMIF-DONES becomes operational, irradiation testing will continue in fission reactors, as their neutron spectra are sufficiently similar for these functional materials to enable validation of high-fluence behavior and temperature dependencies.

These materials closely interface with other EUROfusion work packages. In diagnostics, optical components support plasma monitoring via spectroscopy and interferometry, where neutron damage can compromise signal accuracy. For H&CD, dielectric windows in ECRH and ICH systems must comply with RIC and tan???? limits. Integration with breeding blankets and plasma also requires addressing cross-effects like tritium permeation/retention. Ongoing collaborations aim to optimize material designs for reliable long-term fusion performance.

# Interlaboratory study on miniaturized tensile specimens in the frame of EUROfusion programme

Aleksandr ZINOVEV (Institute for Nuclear Energy Technology, Belgian Nuclear Research Centre (SCK CEN))

Francisco MIRANDA (Institute of Mechanics, Materials and Civil Engineering (iMMC), Université catholique de Louvain (UCLouvain); Institute for Nuclear Energy Technology, Belgian Nuclear Research Centre (SCK CEN))

Khurram AMJAD (United Kingdom Atomic Energy Authority, Culham Centre for Fusion Energy, Culham Science Centre)

David CINGER (HUN-REN Centre for Energy Research, Laboratory of Fuel and Reactor Materials)

Rafał MOLAK (Faculty of Materials Science and Engineering, Warsaw University of Technology)

Ermile GAGANIDZE (Karlsruhe Institute of Technology, Institute for Applied Materials)

Chih-Cheng CHANG (Institute for Nuclear Energy Technology, Belgian Nuclear Research Centre (SCK CEN))

Rebeca HERNÁNDEZ (División de Materiales de Interés energético, CIEMAT)

Marta SERRANO (División de Materiales de Interés energético, CIEMAT)

Rafael GALLEGO (Department of Structural Mechanics and Hydraulic Engineering, University of Granada)

Dmitry TERENTYEV (Institute for Nuclear Energy Technology, Belgian Nuclear Research Centre (SCK CEN); Department of Materials, Textiles and Chemical Engineering, Ghent University)

Gerald PINTSUK (Forschungszentrum Juelich GmbH, Institute for Energy and Climate Research – Plasmaphysics (IEK-4))

Giacomo AIELLO (EUROfusion PMU)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Structural materials used in the first wall, divertor and blanket of a fusion reactor (such as steels, tungsten and copper alloys) will be exposed to harsh operational conditions, which can cause degradation of their mechanical properties. Evaluating these properties, including in the as-irradiated state, is indispensable for a reliable design of fusion facilities. Due to the space limitations of irradiation facilities, the cost of irradiation campaigns and the technical challenges of ensuring the uniformity of irradiation conditions of large specimens, it is essential to develop Small Specimens Testing Techniques (SSTT), which provide reliable results, comparable to those obtained on standard specimens. To this end, small specimens must retain key characteristics of standard specimens, e.g., a cross-section containing a sufficient number of material's grains, as well as comparable deformation modes and stress distribution similar to those expected in standard size specimens. Typical thickness of miniaturized tensile specimens used in Europe and the USA is 0.75–1 mm. Importantly, in order to enable a material qualification campaign relying on small specimens in accordance with the requirements of design codes such as RCC-MRx, specific features related to testing of small specimens must be included in international standards for mechanical tests, such as ASTM or ISO.

In this work, we present the progress achieved by the EUROfusion SSTT group within the Work Package MATerials (WPMAT) toward this goal. Interlaboratory studies were carried out on miniaturized tensile specimens to quantify precision and bias for standardization based on statistical analysis of the results in accordance with ASTM E691. In the first study, one material (EUROFER97) was tested at room temperature and elevated temperature using three small specimen geometries including the SS-J3. In the second study, three materials (including the European reference Nimonic 75 BCR-661B) were tested at RT using two small specimen geometries. Several tests on standard-size specimens were performed as reference. Precision and bias statements were made based on intralaboratory statistical consistency k, interlaboratory statistical consistency h, as well as repeatability %r and reproducibility %R values.

# The concrete shielding experiment at the Frascati Neutron Generator

Davide FLAMMINI (ENEA)
Andrea COLANGELI (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy )
Fabio MORO (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy )
Alberto PREVITI (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy )
Marta DAMIANO (University of Rome Tor Vergata, Department of Industrial Engineering, Rome, Italy)
Roberto LICCIONE (University of Rome Tor Vergata, Department of Industrial Engineering, Rome, Italy)
Virginie LOMBARDI (Università della Sapienza, Roma, Italy )
Stefano LORETI (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy )
Michele LUNGARONI (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy )
Guglielmo PAGANO (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy )
Antonino PIETROPAOLO (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy )
Simone NOCE (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy )
Rosaria VILLARI (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy )

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Concrete represents one of the primary shielding materials for future nuclear fusion facilities owing to its cost-effectiveness, structural properties, and capability to attenuate neutrons from D-T reactions through moderation and capture. Accurate modelling of neutron transport in thick concrete structures is therefore essential for the validation of nuclear data libraries and transport codes employed in the design and safety assessment of fusion reactors. While concrete shielding has been extensively investigated in fission environments, integral experimental data under fusion-relevant 14 MeV neutron fields remain comparatively scarce. To address this gap, two major experiments, supported by EUROfusion, were performed in 2023 at the Nuclear Physics Institute of the Czech Academy of Sciences and in 2025 at the Frascati Neutron Generator (FNG), employing the same set of concrete slabs. The focus of this work is the FNG experiment, that consisted of thirteen concrete slabs (50 × 50 × 5 cm3) arranged in a stacked configuration and positioned at 5.3 cm from the tritiated target, in which the fusion neutron are generated after collision of the accelerated deuteron ions. Activation foils were used to characterize neutron transport and spectral evolution within the assembly. Seven materials (Al, Nb, Ni, Fe, In, Au, and W) were selected to sample neutron-induced reactions sensitive to different energy regions, spanning from thermal energies up to 14 MeV. The foils were positioned at five depths inside the concrete mock-up, ranging from approximately 10 cm to 50 cm, and were irradiated during three dedicated campaigns. Induced activities were measured by means of a high-purity germanium (HPGe) detector, and experimental reaction rates were derived from gamma spectrometry analysis including efficiency calibration and decay corrections. A detailed Monte Carlo model of the full experimental configuration was developed using MCNP. The model includes the FNG bunker hall, the terminal section of the deuteron accelerator with the tritiated target, and all relevant geometrical details of the concrete mock-up. Neutron transport calculations were performed using the JEFF-4.0 and FENDL-3.2 nuclear data libraries, while IRDFF-II dosimetry cross sections were adopted for activation reactions. Calculated-to-experimental (C/E) reaction rate ratios were evaluated as a function of depth to assess the predictive capability of current nuclear data and transport modelling approaches for thick concrete shielding under fusion-relevant irradiation conditions.

# Creep Assessment of CuCrZr Alloys and Pure Copper tubular samples

Luca FERRANTE (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy )
Annunziata Anna SATRIANO (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy )
Andrea MORIANI (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy )
Selanna ROCCELLA (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy )
Pierdomenico LORUSSO (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy )
Jeong-Ha YOU ( Max-Planck-Institute for Plasma Physics (IPP), D-85748 Garching bei Munchen)

## Topics

J. Materials Technology

## Communication type

Poster

## Abstract

Understanding creep behaviour is essential for applications in which materials are exposed to sustained mechanical loads at elevated temperatures, such as plasma-facing components in fusion reactors. In divertor target assemblies, pure copper and CuCrZr alloy are widely used due to their excellent thermal properties, although their performance can be significantly affected by time-dependent deformation phenomena.

To investigate this aspect, a research activity has been carried out at ENEA within the framework of the EUROfusion programme dedicated to divertor target materials for fusion power plant applications. A reliable thermo-mechanical characterization of these materials is crucial for their safe and effective use in in-vessel components.

This study presents the experimental characterization of the creep behaviour of CuCrZr alloy and pure copper at temperatures up to 600 °C. The objective is to evaluate their time-dependent deformation under constant tensile load and to determine the parameters governing steady-state creep. Unlike conventional creep studies performed on standard specimens, the tests were carried out on tubular samples representative of ITER-like divertor cooling tubes, ensuring realistic stress distribution and boundary conditions.

The material constants of the Norton–Bailey law were determined through linear regression of the steady-state creep data in logarithmic form. These parameters provide the basis for describing the stress dependence of creep rate and are essential for predictive modelling. A dedicated high-temperature testing configuration was adopted, including custom-designed gripping systems developed to ensure stable load transfer and prevent slippage under combined thermal and mechanical stresses.

Failure was observed only under the highest applied stress levels, while in the remaining cases the tests were interrupted after achieving stable secondary creep conditions. The results highlight the strong dependence of creep rate on both stress and temperature and provide quantitative data for model calibration.

The ultimate goal of this work is to establish a consistent experimental database suitable for accurate finite element modelling (FEM) of joining processes, such as diffusion bonding, involving copper-based materials. The outcomes support the development and reliability assessment of divertor target components for future fusion power plants.

# Advanced Multilayer Coatings for Tritium Permeation Mitigation and Corrosion Protection in WCLL PbLi Environments

Massimo Emilio ANGIOLINI (ENEA - Italian National Agency for New Technologies, Energy and Sustainable Economic Development)

### Topics

J. Materials Technology

### Communication type

Oral


### Abstract

To minimize fuel losses and prevent material degradation, the Water-Cooled Lithium Lead (WCLL) breeding blanket of the EU-DEMO requires effective tritium permeation barriers. These barriers are essential to limit tritium accumulation within reactor walls and to prevent its permeation through structural components. At the same time, exposure to liquid lead–lithium (PbLi) can severely compromise structural integrity and mechanical properties through dissolution, intergranular penetration, and interstitial mass transfer. A key strategy to address both challenges is the application of protective coatings to breeding blanket structures and coolant systems in direct contact with PbLi.

In this work, a multilayer architecture is proposed to combine the corrosion resistance of refractory metals with the hydrogen permeation resistance of oxide layers, thereby achieving both high tritium diffusion barrier efficiency and strong corrosion resistance in PbLi environments. The developed systems consist of alternating layers of niobium (Nb) and oxides, specifically $Nb_2O_5$, and $Y_2O_3$, forming $Nb/Nb_2O_5$ and $Nb/Y_2O_3$ multilayer coatings.

The multilayer design provides enhanced structural resilience. By integrating complementary materials and engineered interfaces, it mitigates multiple failure modes, interrupts defect propagation, and hinders crack growth. It also improves adhesion and mechanical compatibility with the steel substrate, introducing redundancy so that localized defects do not compromise the overall barrier performance. Furthermore, the multiple interfaces act as effective obstacles to hydrogen transport by disrupting lattice continuity, creating trapping sites, and lowering the effective diffusion coefficient, thereby reducing overall permeation.

The coatings were fabricated using the PVD-DIBS technique, producing stacks of four alternating metal and oxide layers. In each configuration, the oxide layer was placed adjacent to the steel substrate to serve as a diffusion barrier, while the outer Nb layer provided protection against PbLi corrosion.

This study reports the development and comprehensive evaluation of these advanced multilayer coatings for PbLi applications, including detailed microstructural characterization, hydrogen permeation testing, and corrosion assessments in liquid lead–lithium.

# Codification of Korean Reduced Activation Ferritic/Martensitic Steel, ARAA, in RCC-MRx

Yi-Hyun PARK (Korea Institute of Fusion Energy)
KIM SUNGYU (Korea Institute of Fusion Energy (KFE))
GWON HYOSEONG (Korea Institute of Fusion Energy (KFE))
AHN MU-YOUNG (Korea Institute of Fusion Energy (KFE))
KIM WOONG CHAE (Korea Institute of Fusion Energy (KFE))
CHUN YOUNG-BUM (Korea Atomic Energy Research Institute (KAERI))
LEE HYEONG-YEON (Korea Atomic Energy Research Institute (KAERI))

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Since 2012, the Korea Institute of Fusion Energy (KFE), in collaboration with the Korea Atomic Energy Research Institute (KAERI), has conducted an R&D program to develop a Korean Reduced Activation Ferritic/Martensitic (RAFM) steel as a structural material for in-vessel components of fusion energy devices. The developed RAFM steel has been designated as ARAA (Advanced Reduced Activation Alloy). ARAA contains 0.01 wt.% zirconium to improve creep strength and impact resistance, which distinguishes it from conventional RAFM steels. Industrial-scale heats of ARAA, ranging from 3 to 6 tons, have been produced by Korean steel manufacturers to verify reproducibility under identical processing routes and across different manufacturers. More than three industrial-scale heats with thickness ranging from 20 mm to 60 mm were utilized to establish a comprehensive material properties database for codification in RCC-MRx.

All material tests were conducted by ISO 17025-certified testing agencies in accordance with EN ISO and ASTM standards. The database includes physical, thermal, and basic mechanical properties such as tensile and low-cycle fatigue behavior, statistically evaluated using test results up to 700 oC. Based on these data, design-relevant material parameters have been derived and incorporated into RCC-MRx Section IV (2025 Edition). In addition, the fabrication procedure, testing methods, and requirements for product specifications (RPS) including required chemical composition, microstructure, and mechanical properties criteria, have been formally documented. The codification of ARAA in RCC-MRx provides a regulatory and design framework for application in fusion energy devices.

Time-dependent properties, including creep and thermal aging, as well as environmental properties such as neutron irradiation and material compatibility, are currently under investigation. These data will be progressively implemented into RCC-MRx to further support the qualification of ARAA as a structural material for in-vessel components of fusion energy devices.

# Mechanical response to microscale compression of tungsten to self-irradiation and helium implantation

Anthony FLAMENT (CEA IRFM)
Mounir ALFAZZAA (CEMHTI CNRS Orléans)
Zhiwei HU (CEMHTI CNRS Orléans)
Alan DURIF (CEA - IRFM)
Szilvia KALÁCSKA (Laboratoire Georges Friedel - Mines de Saint-Etienne)
Guillaume KERMOUCHE (Laboratoire Georges Friedel - Mines de Saint-Etienne)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Tungsten is commonly employed as plasma facing material due to its high melting point (3422 °C) and low hydrogen isotope chemical affinity. Under ITER/DEMO environment (high neutron bombardment, ion fluxes and heat loads), thermomechanical properties of tungsten can be altered and reduce the PFC lifetime. Self-irradiation & Helium beam implantation are commonly used to simulate fusion neutron damage and Helium implantation in tungsten, notably to study the evolution of its microstructure approaching ITER/DEMO relevant conditions. However, self-irradiation & Helium implantation modify tungsten microstructure on a few micrometers deep layer, necessitating the use of dedicated micro-scale characterization methods such as micro-pillar compression to characterize the thermomechanical properties of the aged near surface. Micropillar compression tests were therefore conducted on well-characterized samples (mechanically polished & annealed monocrystalline & polycrystalline tungsten samples) to assess the sensitivity of the method to self-irradiation (W+ ions, 20 MeV, 0.1 dpa, Room Temperature) and Helium beam implantation (He+ ions, 0.5-1.5 MeV, 2-3x1020 He.m-2, Room Temperature). Electron backscatter diffraction mapping was performed to confirm the initial crystallographic orientation of the characterized area prior to focused ion beam micropillar machining. The room temperature plastic behavior of self-irradiated and Helium implanted micropillars is reported and compared to the typical macroscale results exhibiting brittle failure upon loading. The obtained mechanical data indicate sensitivity to heavy-ion irradiation at 0.1 dpa and to Helium implantation at a fluence of 3x1020 He/$m^2$. The results are correlated to Doppler Broadening Positron Annihilation Spectroscopy measurements which show self-irradiated defects are larger than monovacancies.

# Effect of Post-Heat-Treatment Conditions on Mechanical Properties of Hot-pressed Diffusion-bonded ARAA

Sungyu KIM (Korea Institute of Fusion Energy)
Hyoseong GWON (Korea Institute of Fusion Energy)
Yi-Hyun PARK (Korea Institute of Fusion Energy)

### Topics

J. Materials Technology

### Communication type

Poster

### Abstract

In-vessel components of fusion energy devices require internal cooling channels for heat removal. Diffusion bonding is considered a promising technique for fabricating such cooling channels. For structural reliability, diffusion-bonded joints must exhibit mechanical properties comparable to the base metal. In this study, the effects of post-heat-treatment conditions on hot-pressed diffusion-bonded ARAA were investigated to achieve mechanical properties comparable to those of the base metal.

Two ARAA plates were diffusion bonded by hot pressing at 1150 °C for 3 h under 12.5 MPa. The bonded plate was subsequently heat-treated by normalizing at 1000 °C for 40 min followed by tempering at 750 °C for 70 min. However, the Charpy absorbed energy of the bonded joint decreased from 294 J for the base metal to 240 J. Extending the tempering time to 2 h did not improve the mechanical properties. In contrast, applying double normalizing and double tempering increased the Charpy absorbed energy of the bonded joint to 306 J. The yield strength and tensile strength were 503 MPa and 621 MPa, respectively, which were comparable to 496 MPa and 626 MPa for the base metal. However, the prior austenite grain size remained approximately at ASTM Grain Size Number 8 under all post-heat-treatment conditions. In this presentation, the mechanism responsible for the recovery of mechanical properties through the double normalizing and double tempering process will be discussed based on microstructural analysis.

# Characterization of W/EUROFER functionally graded coatings for fusion applications

Siyu ZHANG (Karlsruhe Institute of Technology, Institute for Applied Materials)
Ashwini Kumar MISHRA (Karlsruhe Institute of Technology (KIT), Institute for Applied Materials, 76344 Eggenstein-Leopoldshafen, Germany)
Jarir AKTAA (Karlsruhe Institute of Technology (KIT), Institute for Applied Materials, 76344 Eggenstein-Leopoldshafen, Germany)

### Topics

J. Materials Technology

### Communication type

Poster

### Abstract

The First Wall is a vital component for future DEMO fusion reactors. To protect its structural material, the EUROFER steel, from high heat flux and heavy particles escaping the plasma, tungsten has been selected for a plasma-facing coating due to its high melting point, high sputtering resistance, and high thermal conductivity. However, the mismatch in the coefficient of thermal expansion (CTE) between tungsten and EUROFER can lead to delamination during the coating process and thermal load in fusion operation. A functionally graded material (FGM) layer between the top tungsten and the steel substrate offers a solution to this issue.

The FGM layer, consisting of 5 interlayers with tungsten volume fraction, increasing from 25% to 75%, and the tungsten top coating were deposited by low-pressure plasma spraying (LPPS). Comprehensive mechanical and microstructural characterizations were then performed. For mechanical characterization, micro-indentation tests were performed on the top tungsten coating, interlayers and in the steel substrate, starting from the interface. To determine the fracture toughness of W/EUROFER FGM coating, bending tests were performed. Samples were tested with varying notch lengths to evaluate the effect of the interlayers on fracture toughness. For microstructural characterization, SEM and image threshold analysis were conducted to quantify the layer thickness, porosity, and actual tungsten content in different regions of the plasma-sprayed coating.

To determine the elastic-plastic properties of the interlayers, slices were cut using electrical discharge machining (EDM), and a standard grinding and polishing procedure was followed to isolate the individual interlayers. Dog bone-shaped micro-tensile specimens were extracted from the individual interlayers by laser cutting, and subsequent tests were performed under vacuum using a customized high-temperature micro tensile and micro creep testing machine developed in-house. The machine consists of a furnace with a max. temperature of 1100 °C, an actuator with max. load of 200 N and max. travel of 40 mm. The load applied is measured with a load cell, while the displacement is measured using a digital image correlation (DIC) system enabling in-situ strain measurement.

The obtained results will be implemented in subsequent thermal-mechanical FEM simulations, helping the service lifetime assessment of W/EUROFER FGM coating under fusion-relevant thermal cycle conditions.

# Advancement of functionally graded W/steel coatings for DEMO and ITER First Wall applications

Ashwini Kumar MISHRA (Karlsruhe Institute of Technology (KIT), Institute for Applied Materials, Eggenstein-Leopoldshafen, Germany)
Siyu ZHANG (Karlsruhe Institute of Technology (KIT), Institute for Applied Materials, Eggenstein-Leopoldshafen, Germany)
Thilo GRAMMES (Karlsruhe Institute of Technology (KIT), Institute for Applied Materials, Eggenstein-Leopoldshafen, Germany)
Arkadi KRETER (Forschungszentrum Jülich GmbH, Institute of Fusion Energy and Nuclear Waste Management – Plasma Physics, Jülich, Germany)
Jan Willem COENEN (Forschungszentrum Jülich GmbH, Institute of Fusion Energy and Nuclear Waste Management – Plasma Physics, Jülich, Germany)
Jeong-Ha YOU (Max-Planck-Institut für Plasmaphysik, Garching, Germany)
Jarir AKTAA (Karlsruhe Institute of Technology (KIT), Institute for Applied Materials, Eggenstein-Leopoldshafen, Germany)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Tungsten coating is a promising candidate as a protective coating for the structural steel of the fusion First Wall. Tungsten and steel, however, have a mismatch in their thermal expansion, which induces high thermal stresses during coating and operational thermal cycles as well. A functionally graded coating was deposited between the top W coating and the steel substrate to mitigate this problem. This work shows the development of W/EUROFER functionally graded material (FGM) coating for the DEMO First Wall and W/SS 316L FGM coating for the temporary ITER First Wall.

Low-pressure plasma-sprayed W/EUROFER FGM coatings were deposited over a steel substrate. Coatings of small to large-scale components were developed. Afterwards, the coating was extended to curved mock-up and prototype-size components. Detailed structural and mechanical characterizations, including microstructure analysis, thickness uniformity measurement, W content analysis, hardness, and fracture mechanical testing, were performed. Furthermore, the phased array ultrasonic testing method was used to analyze the interface characteristics of the coated components. All these characterizations showed that the coatings were deposited homogeneously while maintaining similar coating quality. A plasma exposure test was performed on the coatings, which revealed that the coating's properties with respect to plasma erosion and deuterium retention were similar to those of bulk tungsten. In a further step, a castellated W/EUROFER FGM-coated component was successfully developed. Castellation of coatings showed a reduction in bending and stress in the coated components based on finite element simulations and experiments.

W/SS 316L FGM coatings development work was carried out as a part of the EUROfusion R&D contributions to support the ITER temporary first wall design activity. The technology options considered and being developed in this framework included thermal spraying at three different processing conditions, namely, vacuum/low-pressure (this work), inert gas/high-pressure and cold spraying. Focus was placed on industrial upscaling with optimised quality features.

Finite element simulations were performed considering the deposition condition and the First Wall relevant heat flux conditions. A four-interlayer coating system with an FGM thickness of 0.8 mm was the optimal condition identified based on simulations. Low-pressure plasma-spray W/SS 316L FGM coatings were deposited using the optimized parameters over small-scale steel plates. Detailed structural and mechanical characterizations were performed, which confirmed established specifications and quality. Afterwards, medium-scale and large-scale steel plates were coated using the established parameters.

This work demonstrates the feasibility of depositing functionally graded coatings over steel in sizes ranging from small to application relevant.

# Structural integrity assessment of HCPB breeding blanket against exhaustion of ductility using the EU-DEMO Design Criteria for In-Vessel Components

Vishnu AMBIKADEVI (UK Atomic Energy Authority (UKAEA))
Guangming ZHOU (Institute for Neutron Physics and Reactor Technology (INR), Karlsruhe Institute of Technology (KIT))
Yiqiang WANG (UK Atomic Energy Authority (UKAEA))

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

The Helium Cooled Pebble Bed (HCPB) is one of the breeding blanket concepts under investigation by EUROfusion for the EU DEMO fusion reactor, with EUROFER97 as the proposed baseline structural material. Structural integrity assessments during the concept design phase were primarily based on the RCC-MRx elastic design-by-analysis approach. However, to reduce over-conservatism and enhance performance by employing an inelastic design-by-analysis approach based on advanced non-linear finite element analysis, the DEMO Design Criteria for In-Vessel Components (DDC-IC) is currently being developed within EUROfusion. Therefore, in this study, we performed the structural integrity assessment of the HCPB breeding blanket conceptual design based on the newly developed design rules within the DDC-IC, particularly against the exhaustion-of-ductility failure mode, aiming to form a basis for future component-level assessments employing inelastic analysis and to investigate the effects of joints/welds. The systematic region-wise assessment results revealed several hotspots where the finite element strain prediction exceeds the allowable limits, including the joints between the first wall and pressure tubes, pressure tubes and breeding zone backplate, and pressure tubes and breeder pin assembly. Another key outcome from the inelastic stress analysis is that the failure surface needs to depend on both triaxiality and the Lode parameter, as their ranges observed in the breeding blanket are often quite wide, consistent with the current developments within the DDC-IC. These results will feed into the development of the DDC-IC to establish a more representative testing matrix for the validation of design criteria. Additionally, a sub-modelling approach was developed to perform high-fidelity analyses of the regions of interest containing joints, with advanced material constitutive models more efficiently. This approach will be used to assess welds and joints in future structural integrity assessments of the breeding blanket, with more accurate modelling of geometries as well as material behaviour. All these results will support the overall breeding blanket design optimisation by reducing over-conservatism while maintaining the highest quality standards, mitigating mechanical failures and improving its lifetime.

# Structural Materials Development for Fusion Applications at Oak Ridge National Laboratory: Irradiation Campaigns, International Collaborations, and Industrial-Scale Alloy Deployment

Tim GRAENING (ORNL)
Josina GERINGER (ORNL)
Yan-Ru LIN (ORNL)
Yukinori YAMAMOTO (ORNL)
Yanli WANG (ORNL)
Ying YANG (ORNL)
Weicheng ZHONG (ORNL)
Xiang CHEN (ORNL)
TS BYUN (ORNL)
Marie ROMEDENNE (ORNL)
Ezekial UNTERBERG (ORNL)
Yutai KATOH (ORNL)
Steven ZINKLE (University of Tennessee, Knoxville)
Eric LASS (University of Tennessee, Knoxville)
Lance SNEAD (Stony Brook University)
Sara FERRY (Massachusetts Institute of Technology)
Jaime MARIAN (University of California-Los Angeles)
Michael RIETH (Karlsruhe Institute of Technology (KIT))
Carsten BONNEKOH (Karlsruhe Institute of Technology (KIT))
Amy GANDY (United Kingdom Atomic Energy Authority (UKAEA))
Slava KUKSENKO (United Kingdom Atomic Energy Authority (UKAEA))
Dai HAMAGUCHI (National Institute for Quantum Science and Technology (QST))
Takashi NOZAWA (National Institute for Quantum Science and Technology (QST))
Hiroyasu TANIGAWA (National Institute for Quantum Science and Technology (QST))

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

The realization of near-term fusion pilot plants requires structural materials that combine irradiation tolerance, high-temperature mechanical performance, and industrial scalability. This presentation summarizes an integrated structural materials program performed by staff at Oak Ridge National Laboratory (ORNL) and other institutions that couple domestic alloy development with international irradiation collaborations to accelerate materials readiness for fusion energy. The structural materials program integrates irradiation science, international collaboration, industrial manufacturing, and accelerated structural material deployment. Under the umbrella of the DOE-EUROfusion collaboration, irradiation campaigns and post‐irradiation examination (PIE) of Eurofer97 steels have been conducted on samples irradiated up to 20 dpa. Mechanical testing for handbook data generation is performed alongside microstructural investigations. In Phase II, augmentation capsules incorporating isotopically tailored additions are being irradiated in HFIR to reproduce fusion-relevant helium production with PIE results of 30 dpa 54Fe tailored materials shown here. Irradiation effects in UKAEA and ORNL fusion structural materials, including ferritic–martensitic and oxide dispersion strengthened steels, are being investigated through neutron irradiation in HFIR and subsequent PIE at ORNL and UKAEA, providing mechanical and

microstructural data for newly irradiated and archival materials. Lastly, more than 40 years of collaboration between ORNL and the National Institutes for Quantum Science and Technology (QST) has yielded significant progress, including supplementing a database for F82H steel, advancing studies of irradiation effects. Recent accomplishments will also be summarized on precipitate-strengthened structural materials development performed under the DOE FIRE IMPACT Collaborative, which aims to accelerate deployment-ready fusion materials through coordinated alloy development, industrial scale-up, and stepwise code casing. The program includes precipitate strengthened ferritic martensitic steel (castable nanostructured alloy, CNA) along with novel vanadium alloy development as an alternative structural material system. Extensive PIE of CNA variants irradiated in HFIR as part of the fusion materials base program will also be summarized. Laboratory-scale heats and early compositional variants are evaluated through tensile testing, hardness measurements, fracture toughness testing, and high?resolution microstructural analysis to quantify irradiation-induced hardening, stability of nanoscale precipitates, and defect evolution. Ongoing material development efforts on proposed reduced-activation bainitic-ferritic steels applicable for vacuum vessels and structural rings will be summarized. These efforts focus on improving the fabricability of large-scale fusion structural components by eliminating the need for post-weld heat treatments (PWHT). Compositional modification of 3Cr-based bainitic-ferritic steel developed in 1990s at ORNL has successfully preserved high-temperature strength and room-temperature toughness without PWHT, while maintaining the producibility of both the material and its weldments.

# Helium-Induced Blistering and Subsurface Bubble Evolution in ITER-Grade and K-Doped Tungsten at Elevated Temperature

JunHui HAN (Myongji University)
Woohyun LIM (Myongji University)
Sangeun KIM (Myongji University)
Hyung-Ha JIN (Korea Atomic Energy Research Institute (KAERI))
Sanghoon NOH (Pukyong National Unversity)
Kwon SUNGJIN (Korea Institute of Fusion Energy (KFE))
Dong Won LEE (Korea Atomic Energy Research Institute (KAERI))
Chansun SHIN (Myongji University)

### Topics

J. Materials Technology

### Communication type

Poster

### Abstract

Tungsten is a leading plasma-facing material for fusion divertors, yet helium exposure can induce near-surface blistering and subsurface bubble evolution that may degrade thermo-mechanical integrity and affect hydrogen isotope retention. Here we compare helium-driven surface damage and subsurface bubble formation in polycrystalline ITER-grade tungsten (IGW) and potassium-doped tungsten (KdW) under elevated-temperature implantation conditions.

For implantation-driven surface blistering, IGW and KdW specimens were irradiated with 20 keV He ions at 700 °C to a fluence of $1\times10^{18}$ ions/cm$^2$. SRIM calculations indicate that implanted He and displacement damage are confined to the near-surface region (tens of nanometers), consistent with an implantation-like blistering regime. Post-irradiation SEM revealed dome-shaped blisters, partially opened blisters, and crater-like features associated with cap rupture/exfoliation. A half-mask geometry enabled within-sample comparison of irradiated and unirradiated regions. IGW showed pronounced spatial heterogeneity: blister-rich zones formed band-like patterns with widths of several tens of micrometers, resembling the rolling-induced banded microstructure observed by EBSD. In contrast, KdW exhibited a more spatially uniform blister distribution. Using a TEM-grid mask for localized irradiation mapping, blister-concentrated regions in IGW were found to correlate with near-(001) orientation bands inferred from adjacent unirradiated EBSD data, suggesting strong orientation dependence. Quantitative crater statistics further indicated smaller crater sizes in KdW than in IGW under identical irradiation conditions, consistent with a possible modification of local He accumulation and bubble growth by K-derived trapping features.

To probe subsurface bubble evolution relevant to swelling, complementary 200 keV He irradiation was performed at 700 °C ($1\times10^{17}$ ions/cm$^2$), followed by TEM observations at selected depths guided by SRIM profiles (peak at several hundred nanometers). Nanometer-scale He bubbles were observed near the predicted peak region, with reduced bubble density toward the tail of the implantation range. The combined dataset highlights how processing-derived microstructure and K-related features can influence both near-surface blistering/rupture and deeper bubble evolution at elevated temperature.

#13713

# Influence of Surface Finish on Helium-Ion Blistering of ITER-Grade and K-Doped Tungsten at 700 °C

Woohyun LIM (Myongji University)
JunHui HAN (Myongji University)
Sangeun KIM (Myongji University)
Hyung-Ha JIN (Korea Atomic Energy Research Institute (KAERI))
Sanghoon NOH (Pukyong National Unversity)
Sungjin KWON (Korea Institute of Fusion Energy (KFE))
Dong Won LEE (Korea Atomic Energy Research Institute (KAERI))
Chansun SHIN (Myongji University)

## Topics

J. Materials Technology

## Communication type

Poster

## Abstract

Surface finishing and machining marks are unavoidable in practical divertor monoblock fabrication and may critically influence helium-induced damage in tungsten. This study examines how surface condition modifies implantation-driven helium blistering in polycrystalline ITER-grade tungsten (IGW) and potassium-doped tungsten (KdW) at elevated temperature.

Specimens were prepared with two representative surface states: (i) electropolished surfaces optimized for EBSD-quality characterization, and (ii) mechanically finished surfaces representative of technical machining, produced by coarse abrasive polishing (e.g., 100–200 grit) that introduces increased roughness and a deformed near-surface layer. Both IGW and KdW were irradiated with 20 keV He ions at 700 °C to $1\times10^{18}$ ions/cm$^2$. SRIM calculations confirm that implanted He and damage are localized within the near-surface region, where surface condition is expected to strongly affect defect sinks, local stress/strain accommodation, and gas release pathways.

Post-irradiation SEM was used to quantify blister statistics (number density, size distribution, area fraction) and rupture behavior (fraction of partially opened blisters and crater-like features). Particular attention was paid to the spatial correlation between surface machining features (e.g., scratches/grooves) and blister nucleation. Preliminary observations indicate that mechanically finished IGW surfaces can exhibit line-guided arrays of small dome blisters along residual scratch lines, implying that surface deformation and stress fields associated with machining may localize helium bubble coalescence. By contrast, electropolished surfaces provide a baseline morphology with reduced topographic artifacts, facilitating direct comparison of intrinsic microstructure effects. For KdW, the role of K-derived features combined with surface finish is evaluated by comparing the uniformity of blister distributions and the propensity for cap rupture under matched irradiation conditions.

The results will be discussed in terms of competing mechanisms of blister growth versus premature rupture/exfoliation, including the effects of near-surface densification/porosity, deformation-induced defect populations, and potential helium release pathways enabled by surface roughness. These findings aim to provide practical guidance for surface-finish control in tungsten plasma-facing components and to define processing windows that mitigate helium-induced surface damage at divertor-relevant temperatures.

# Suppressing Grain Growth in Tungsten up to 2000 °C: The W–W?C Approach

Petra JENUS BELEC (Jožef Stefan Institute)
Aleksander UCAKAR (Jožef Stefan Institute)
Conrad V. HARTUNG (Jožef Stefan Institute)
Andreja SESTAN ZAVASNIK (Jožef Stefan Institute)
Sabina MARKELJ (Jožef Stefan Institute)
Esther PUNZÓN-QUIJORNA (Jožef Stefan Institute)
Borut ZUZEK (Institute of Metals and Technology)
Jaka BURJA (Institute of Metals and Technology)
Marko SEDLACEK (Institute of Metals and Technology)
Elena TEJADO (Centro de Investigación en Materiales Estructurales-CIME. Universidad Politécnica de Madrid)
Sandra TARANCÓN (Centro de Investigación en Materiales Estructurales-CIME. Universidad Politécnica de Madrid)
Jose Ygnacio PASTOR (Centro de Investigación en Materiales Estructurales-CIME. Universidad Politécnica de Madrid)
Marius WIRTZ (Forschungszentrum Juelich GmbH, Institute of Fusion Energy and Nuclear Waste Management - Plasma Physics (IFN-1), 52425 Juelich)
Mauricio GAGO (Forschungszentrum Juelich GmbH, Institute of Fusion Energy and Nuclear Waste Management - Plasma Physics (IFN-1), 52425 Juelich)
Daniel DICKERS (Max Planck Institute for Plasma Physics, Garching)
Johann RIESCH (Max Planck Institute for Plasma Physics, Garching)
Bernd BÖSWIRTH (Max Planck Institute for Plasma Physics, Garching)
Aljaz IVEKOVIC (Jožef Stefan Institute)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Tungsten (W) is the primary candidate for plasma-facing components in the DEMO divertor. However, its intrinsic brittleness, recrystallization-induced degradation, and pronounced grain growth at elevated temperatures limit its performance under fusion-relevant thermal loads. To overcome these drawbacks, W–$W_2C$ composites were developed in which tungsten sub-carbide ($W_2C$) particles are formed in situ at W grain boundaries during Field Assisted Sintering (FAST) [1].

The in situ formed $W_2C$ inclusions effectively pin grain boundaries, suppressing abnormal grain growth even at temperatures up to 2000 °C. At the same time, carbide formation promotes densification and eliminates oxygen from the starting powder without hydrogen treatment. Compared to pure W, isotropic W–$W_2C$ composites show refined microstructures, enhanced flexural strength at room and elevated temperatures, and improved long-term thermal stability. For low $W_2C$ contents (2–4 wt%), the ductile-to-brittle transition temperature (DBTT) ranges between 200–400 °C, comparable to or lower than conventional isotropic tungsten grades. Thermal ageing above 1250 °C for up to seven days confirms the stability of the microstructure due to grain boundary pinning. Deuterium retention measurements further demonstrate acceptable trapping levels under fusion-relevant conditions [2].

Building on these findings, isotropic FAST-sintered W–$W_2C$ composites were subjected to controlled high-temperature plastic deformation (compression and/or forging) to produce anisotropic microstructures with tailored grain morphology. Phase composition (XRD, EBSD), microstructure evolution, and mechanical properties were systematically evaluated before and after additional thermal treatment at 1600 °C for 24 h. Attention was given to verifying whether $W_2C$ inclusions effectively inhibit grain growth also in anisotropic architectures.

The combination of high thermal stability, improved mechanical performance, reasonable thermal conductivity (>100 W/mK at 1000 °C), and controlled hydrogen isotope retention makes W–$W_2C$ composites promising candidate materials for DEMO divertor applications.

References

[1] S. Novak et al, Mater. Sci. Eng. A, 2020, 772, 138666.

[2] P. Jenuš et al., Nucl. Mater., 2023, 581, 0–6.
Acknowledgement: This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 — EUROfusion). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them. This project has received funding from the Slovenian Research Agency (P2-0087 and P2-0405).

#13721

# Vanadium alloy to tungsten armour joining for fusion first-wall/blanket & plasma-facing components

Kamlesh PATEL (University of Birmingham)
Auronadaya BHATTACHARYA (University of Birmingham)
Samara LEVINE (Tokamak Energy Ltd, Milton Park, Oxfordshire OX14 4SD, United Kingdom)
Sandeep IRUKUVARGHULA (Tokamak Energy Ltd, Milton Park, Oxfordshire OX14 4SD, United Kingdom)
Daniel WILKINSON (Materials Department of the University of Oxford)
Andrew SOWDER (Electric Power Research Institute (EPRI), Charlotte, NC, USA)
Marc ALBERT (Electric Power Research Institute (EPRI), Charlotte, NC, USA)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

As the global push for clean energy through fusion takes centre stage extensive worldwide research is focused on finding suitable materials for the first wall/blanket structural materials and its joining with the plasma-facing armour. For Tokamak reactors while W is the primary plasma-facing candidate material for its high melting point and thermal properties; Vanadium alloys (V-4Cr-4Ti) have been identified as leading structural material in liquid lithium (Li) blanket concept due to its good creep thermal properties, fabricability, compatibility with liquid Li and resistance to irradiation damage.

The study aims to investigate the fabrication and interfacial properties of the transition joints of V alloys and W using Directed Energy deposition (DED) – diode laser cladding and diffusion bonding using FAST SPS (Field Assisted Sintering Technique Spark Plasma Sintering) focusing on processing parameters and the effect of microstructure and thermomechanical properties of the joint.

Literature and initial trails highlight concerns with residual stresses because of the rapid thermal cycles. Initial experimental trails indicated that standard laser cladding parameters often induce high tensile stresses at the interface, leading to interfacial cracking or brittle substrate failure. To mitigate the residual stresses, we explored cooling under pressure , preheating and substrate management

Here, we will present a quantitative analysis of the structure-property relationships in the different W-V alloy joints, including a quantification of mechanical properties. Failure modes under preliminary high flux testing will also be presented.

Research jointly sponsored by Tokamak Energy, University of Birmingham and 2024 UKAEA’s FOSTER Fusion Skills Voucher programme.

# Overview of Environmental Strength Properties of Fusion Reactor Structural Materials

Motoki NAKAJIMA (National Institutes for Quantum Science and Technology (QST), Rokkasho Fusion Institute, Japan)
Kato TAICHIRO (National Institutes for Quantum Science and Technology (QST), Rokkasho Fusion Institute, Japan)
Takashi NOZAWA (National Institutes for Quantum Science and Technology (QST), Rokkasho Fusion Institute, Japan)

## Topics

J. Materials Technology

## Communication type

Poster

## Abstract

Fusion reactor materials are subjected to various loadings during actual operation, including self-weight, thermal stresses, and electromagnetic forces. While numerous evaluations of structural material strength properties have been conducted, the acquisition of environmental strength data, where corrosion and loading effects overlap, is not necessarily sufficient. Therefore, it is of significant engineering importance to compile the environmental strength characteristic data obtained to date, clarify the issues, and consider how to incorporate it into standards and codes.

This presentation reviews stress corrosion cracking (SCC) susceptibility evaluation, corrosion fatigue properties, and hydrogen embrittlement susceptibility for reduced activation ferrite/martensitic (RAFM) steel, a blanket structural material, and CuCrZr alloy, a divertor cooling tube material. It identifies challenges for fusion reactors and outlines data acquisition guidelines for future specification and standard development.

For RAFM steel, while the risk of SCC initiation and propagation is low, corrosion fatigue life may decrease to 1/19. Furthermore, all data pertains to the base material. Data is limited for dissimilar welded joints involving RAFM steel and austenitic stainless steel, necessitating expanded experimental data and risk analysis.

For copper alloys, evaluations of environmental assisted cracking (EAC) susceptibility using temperature and dissolved hydrogen concentration as parameters are underway. Under conditions of 230°C and 1.6 ppm dissolved hydrogen addition, distinct EAC occurred. Even under other hydrogen addition conditions, fine crack-like features were observed on the specimen surfaces after testing. After quantifying crack initiation and shrinkage and evaluating the risk of environment-assisted cracking, an EAC map using temperature and dissolved hydrogen concentration as parameters will be created for discussion.

# Deuterium permeation through liquid tin supported in metal mesh structures

Yuya ANDO (Kindai University)
Koki DOI (Kindai University)
Teppei OTSUKA (Kindai University)

## Topics

J. Materials Technology

## Communication type

Poster

## Abstract

It is important for safety to understand the retention and permeation behavior of tritium in materials from the viewpoint of mitigating internal radiation exposure for workers or fuel loss. Liquid tin (Sn) has been proposed as one of the candidates for plasma-facing components or divertor of liquid metals in future fusion reactors due to its low evaporation pressure and easy handling capability. However, the hydrogen permeation behavior in liquid Sn has not been sufficiently revealed and could be much affected by interfacial corrosion at the interface between liquid Sn and supporting metals [Y. Ando, SOFT-2024]. In so-called capillary systems, liquid Sn can be retained within porous media or mesh structures by surface tension. The purpose of this study is to clarify the permeation behavior of hydrogen isotopes through the liquid Sn supported in the metal mesh structures.

A stainless-steel gasket (5 mm thick with a 16 mm diameter hole) was used to hold a structure composed of five stacked molybdenum (Mo) meshes (0.03 mm wire diameter, 80 mesh). Small pieces of Sn rod were melted on the Mo mesh structure. The liquid Sn layer was about 5 mm in thickness, and part of Sn penetrated into the mesh pores. This configuration is referred to as the (Sn)#Mo system. The (Sn)#Mo system separated the two containers: an upstream side for deuterium gas loading and a downstream side for evacuation and measurement of permeated deuterium using a quadrupole mass spectrometer (QMS). Since an advective argon gas flow through liquid Sn in the (Sn)#Mo system was incidentally observed, a palladium (Pd) membrane (0.3 mm thick) was installed beneath the Mo mesh as a safety measure.

Deuterium permeation experiments were conducted at a constant temperature ranging from 473 to 773 K under a constant deuterium pressure ranging from 0.1 to 2.0 kPa. The deuterium permeation flux through the (Sn)#Mo system was one order of magnitude smaller, and its activation energy was lower than that for the bare Pd membrane. This indicated that the permeation behavior was rate-controlled by the (Sn)#Mo system. Overshoot phenomenon were observed during a transient of permeation in each pressure and temperature step though the cause is under investigation. In the presentation, the effect of mesh size controlling tortuosity for liquid Sn will be discussed with respect to interfacial corrosion between the mesh–liquid Sn interface.

# Manufacturing and Analysis of Tungsten Fiber-Reinforced Copper Composite Tubes for Divertor Targets

Pierdomenico LORUSSO (ENEA, Nuclear Department)
Selanna ROCCELLA (ENEA, Nuclear Department)
Davide CAPRINI (ENEA, Nuclear Department)
Marco CEROCCHI (ENEA, Nuclear Department)
Francesco CREA (ENEA, Nuclear Department)
Riccardo DE LUCA (ENEA, Nuclear Department)
Angelo TATÌ (ENEA, Nuclear Department)
Luigi VERDINI (ENEA, Nuclear Department)
Patrick JUNGHANNS (Max Planck Institute for Plasma Physics)
Katja HUNGER (Max Planck Institute for Plasma Physics)
Alexander VON MÜLLER (Max Planck Institute for Plasma Physics)
Jeong-Ha YOU (Max Planck Institute for Plasma Physics)

### Topics

J. Materials Technology

### Communication type

Poster

### Abstract

The technological development of materials for plasma-facing components is a key aspect for the advancement in fusion reactors design. Among the envisioned concepts, tungsten fiber-reinforced copper (Wf-Cu) composite tubes represent an innovative solution for divertor targets, addressing the limitations of conventional ITER-like CuCrZr pipes currently adopted in reference design. These composite tubes provide enhanced mechanical resistance, being less prone to degradation caused by high temperatures and neutron irradiation respect the CuCrZr ones. Furthermore, they allow a reduction in radioactive waste generation, as tungsten exhibits lower activation compared to copper. Currently, the two investigated infiltration procedures for the fabrication of Wf-Cu composite are the copper casting and the galvanic process. The research aims at manufacturing tubes suitable with small and medium scale target mock-up production, while assessing their compatibility with operating conditions of fusion power plant divertor targets. The present work summarizes the recent advancements in the Wf-Cu composite pipes manufacturing by copper casting. The activities have been focused on modifications to the W fiber density and weaving structure, along with the optimization of the casting procedures in order to enhance the Wf-Cu matrix quality, reducing size and number of defects and improving at the same time the uniformity of the internal and external tube surfaces. Furthermore, new samples have been fabricated using alternative materials for the furnace tools. In particular, Al2O3 has been tested as replacement for the graphite in order to eliminate dust-related contamination of the W fiber and to avoid the risk of W-fibers mechanical properties degradation due to C impurities. The mechanical integrity of the composite tubes has been assessed throughout the integration of non-destructive (Ultrasonic Testing) and destructive (metallographic) examination techniques. Moreover, an evaluation of non-destructive examination methods has been carried out, comparing the Ultrasonic Testing with X-ray tomography to assess their effectiveness in characterizing the internal structure of the matrix. The comparison of the examinations proved that both methods are capable to identify porosities inside the matrix with a minimum dimension up to ~0.5 mm for the ultrasounds and ~0.3 mm for the X-rays, making them suitable for the Wf-Cu composite pipes inspection.

#13783

# The experimental validations of key nuclear data and measurement techniques for IFMIF-DONES

Yuefeng QIU (Karlsruhe Institute of Technology)
Frederik ARBEITER (Karlsruhe Institute of Technology (KIT))
Irene ÁLVAREZ (Consorcio IFMIF-DONES España, Granada)
Martin ANSORGE (Nuclear Physics Institute of the Czech Academy of Sciences)
Santiago BECERRIL (Consorcio IFMIF-DONES España, Granada)
Hari CHOHAN (UKAEA - United Kingdom Atomic Energy Authority)
Miguel CARVAJAL (Universidad de Granada)
Daniil KOLIADKO (Karlsruhe Institute of Technology (KIT))
Axel KLIX ( Karlsruhe Institute of Technology (KIT))
Victor LOPEZ (Universidad Nacional de Educación a Distancia (UNED), C/Juan del Rosal 12, Madrid, Spain)
Fernando MOTA (CIEMAT - LABORATORIO NACIONAL DE FUSION)
Tomasz PIOTROWSKI (Warsaw University of Technology)
David JIMENEZ-REY (CIEMAT - LABORATORIO NACIONAL DE FUSION)
Ocean WONG (UKAEA - United Kingdom Atomic Energy Authority)
Rafal PROKOPOWICZ (National Centre for Nuclear Research (NCBJ))
Grzegorz WOJTANIA (National Centre for Nuclear Research (NCBJ))
Monika WIELGOSZ (National Centre for Nuclear Research (NCBJ))
Milan ŠTEFÁNIK (Nuclear Physics Institute of the Czech Academy of Sciences)
Ján KOZIC (Nuclear Physics Institute of the Czech Academy of Sciences)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

IFMIF-DONES is an accelerator-based neutron irradiation facility, providing the necessary irradiation data for the qualification of materials for the DEMO fusion power plant. The facility comprises a deuteron accelerator, a liquid lithium target, and a test cell with the High Flux Test Module (HFTM), which is utilized to irradiate material specimens. Neutrons and gammas are shielded with steel and concrete of the test cell surrounding the test area.

The Self-Powered Neutron Detectors (SPNDs) and activation foils are the baseline neutron measurement techniques for the HFTM. SPNDs have been recently validated at CERN n_TOF, and additional signal-response tests are planned at the NPI cyclotrons. New foil materials in addition to Fe, Ni, Y, Co, and Au will be proposed, and further irradiations and spectrum-unfolding are planned at the Tohoku University cyclotrons. In parallel, activation of EUROFER samples and the correlation with damage dose has recently been proposed as a feasible approach for DPA estimation. This method has been recently validated at the NPI U-120M cyclotron, and additional validation experiments are planned with the activation foil irradiations.

In addition, key nuclear data validations have been carried out for safety and source-term analysis related to d–Li activation and the production of Be-7 and tritium. Irradiation experiments on Li and LiF targets with deuteron energies up to 40 MeV were done at the GANIL/NFS facility in September 2025, and the results will soon become available. To characterize shielding concrete, experiments were conducted at the NPI cyclotron using a concrete mock-up manufactured with aggregates representative of the DONES material composition. New experiments are planned at MARIA reactor at NCBJ(Poland) to test activation, structural, and nuclear properties of neutron-irradiated concretes. Within the collaboration with QST under the Broader Approach Phase II (BA-II), validation of the current accelerator-neutronics simulation tools was performed through comparison with measurement data obtained using several techniques, including activation foils. This work aims to present a summary of the progress and key findings from these activities.

# IFMIF-DONES Central Instrumentation and Control Systems: Design Consolidation and Recent Advancements within the EUROfusion WPENS Framework

Mauro CAPPELLI (ENEA)
Enrico BOTTA (Ansaldo Nucleare S.p.A)
Javier CRUZ (University of Granada)
Eusebio AL-SOLIMAN (University of Granada)
Bernardo CARVALHO (IPFN, Instituto Superior Técnico, Universidade de Lisboa, Av. Rovisco Pais, 1049-001 Lisboa, Portugal )
Zhe CHEN (Aalborg University, Department of Energy Technology, Aalborg)
Miguel DAMAS (University of Granada)
Jacopo DELFINI (Ansaldo Nucleare S.p.A)
Dawid DWOJEWSKI (S2Innovation)
Victor GUTIERREZ (CIEMAT)
Jorge SOUSA (IPFN, Instituto Superior Técnico, Universidade de Lisboa, Av. Rovisco Pais, 1049-001 Lisboa, Portugal )
David JIMENEZ (CIEMAT)

### Topics

J. Materials Technology

### Communication type

Poster

### Abstract

The International Fusion Materials Irradiation Facility-DEMO-Oriented Neutron Source (IFMIF-DONES) is a high-intensity accelerator-based neutron source designed to qualify structural and functional materials for future fusion reactors. By accelerating deuterons up to 40 MeV onto a liquid lithium target, the facility generates a neutron spectrum representative of DEMO-class fusion power plants. The technological complexity of this infrastructure, combined with stringent safety and availability requirements, demands a highly reliable and fully integrated control architecture. This contribution presents the transition of the IFMIF-DONES Central Instrumentation and Control System (CICS) into its definition design phase. The CICS integrates three core subsystems: CODAC (Control, Data Acquisition, and Communication), responsible for plant-wide supervision, coordination, and data management; the Machine Protection System (MPS), providing fast fault detection and mitigation to prevent equipment damage; and the Safety Control System (SCS), implementing safety functions to protect personnel and the environment under normal and postulated accident conditions. The proposed architecture follows a hierarchical and distributed approach, ensuring scalability, deterministic communication, functional segregation, and high availability through redundancy and fault-tolerant strategies. Recent developments include the assessment of advanced control frameworks for CODAC, refinement of the MPS response architecture to meet stringent reaction-time requirements, and the consolidation of the safety lifecycle and hazard mitigation methodology within the SCS. The CICS design here presented has been developed within the EUROfusion Work Package Early Neutron Source (WPENS). These advancements establish a robust technological foundation for IFMIF-DONES, contributing to the reliable and safe operation of a key facility for the advancement of fusion materials research.

# Effect of Electron Beam Welding and Post?Weld Heat Treatment on the Mechanical and Microstructural Properties of Reference and Modified EUROFER97 Grades

Sahil VALIYEV (Institute for Nuclear Energy Technology, Belgian Nuclear Research Centre (SCK CEN))
Athina PUYPE (OCAS NV)
Carsten BONNEKOH ( Karlsruhe Institute of Technology (KIT))
Serguei GAVRILOV (Institute for Nuclear Energy Technology, Belgian Nuclear Research Centre (SCK CEN))
Dmitry TERENTYEV (Institute for Nuclear Energy Technology, Belgian Nuclear Research Centre (SCK CEN))
Wim DE WAELE (Ghent University)
Roumen PETROV (Ghent University)

### Topics

J. Materials Technology

### Communication type

Poster

### Abstract

EUROFER97, a reduced activation ferritic–martensitic (RAFM) steel developed in Europe, has been selected as the primary structural material for the fabrication of tritium breeding blankets in fusion reactors. To date, four industrial batches have been produced, and EUROFER97 remains the only RAFM steel included in the RCC MRx design code under the probationary phase rules. However, standard EUROFER97 exhibits significant irradiation induced hardening and embrittlement at operating temperatures of 280–325 °C, which correspond to water-cooled tritium breeding blanket concept. To address this limitation, new grades of EUROFER97 were developed by SCK CEN and produced by OCAS NV based on an optimized thermomechanical processing route designed to achieve a refined grain structure and specific precipitation state. These modified grades exhibit a lower ductile-to-brittle transition temperature (DBTT) under both unirradiated and irradiated conditions, without compromising the tensile strength relative to the reference EUROFER97. Another key challenge for breeder blanket fabrication is the weldability of EUROFER97, as the complex geometry of the blanket modules requires multiple welding operations during manufacturing and assembly. Several welding techniques, including fusion and diffusion welding, are under consideration for these purposes. Among fusion welding processes, electron beam (EB) welding offers advantages such as low heat input, narrow heat affected zone, high depth-to-width ratio of fusion zone, and limited distortion as compared to conventional Tungsten Iner Gas (TIG) welding. In this work, EB welding was adopted to produce the joints of the modified and reference EUROFER97 grades. To assess the performance of the joints produced from the modified grades relative to the reference EUROFER97, EB welds were made on 10 mm thick plates and subjected to several post weld heat treatment (PWHT) cycles. These PWHT conditions were designed to optimize the mechanical properties, which are strongly governed by the microstructural changes occurring during the welding thermal cycle. The optimal PWHT parameters were identified based on hardness measurements. Mechanical properties of the joints were evaluated using cross weld (CW) subsize tensile specimens as in welded and PWHT states. Microstructural characterization was performed using light optical microscopy (LOM), and the fracture surfaces of ruptured samples were examined by scanning electron microscopy (SEM).

# Microstructure and mechanical properties of the base metal and TIG weld of advanced reduced-activation alloy and the effect of Zr addition

Min-Woo KANG (Changwon National University)
Joonoh MOON (Changwon National University)
Yi-Hyun PARK (Korea Institute of Fusion Energy (KFE))

## Topics

J. Materials Technology

## Communication type

Poster

## Abstract

Recently, extensive research is being conducted on the development of fusion energy device as a future energy source. The breeding blanket, a key component of fusion energy device, converts the kinetic energy of neutrons generated by fusion reaction into thermal energy. Reduced-activation ferritic-martensitic (RAFM) steels, with their excellent mechanical properties and weldability, are being extensively investigated as a promising candidate material for the breeding blankets. Advanced reduced-activation alloy (ARAA), the Korean RAFM steel, has been reported to exhibit superior creep and impact properties compared to conventional RAFM steel owing to the effect of Zr addition.

In this study, we investigated the effects of Zr addition on the microstructure evolution and mechanical properties in the base metals and tungsten inert gas (TIG)-welded joints of ARAA. For this end, three ARAA plates with different Zr contents were manufactured through vacuum induction melting (VIM) and hot-rolling processes, and filler wires with the same composition as the base metals were prepared for TIG welding. After TIG welding, a post-weld heat treatment (PWHT) was performed at 730 ℃ for 1h. The microstructure of the base metals and TIG welds were observed using scanning electron microscopy (SEM), electron backscattered diffraction (EBSD), and transmission electron microscopy (TEM), and atom probe tomography (APT) analyses was conducted to identify nano-sized MX precipitates. Both the base metals and the TIG welds consisted of typical tempered martensite microstructure, and the welds had relatively large grain sizes compared to the base metals. APT analyses indicates that Zr contributed to the formation of nano-sized MX precipitates. Next, the mechanical properties of the base metals and TIG welds were evaluated by hardness, tensile, and low cycle fatigue tests. The results showed that tensile strength and fatigue life at both room and high temperatures improved with increasing Zr contents. Finally, the correlation between microstructural changes by adding Zr and mechanical properties was discussed.

#13916

# Irradiation Effects on Interfacial Properties of Tungsten-CuCrZr Divertor Monoblock Joints Under Ion Irradiation

Hyung-Ha JIN (Korea Atomic Energy Research Institute)
Sun-Young PARK (Korea Atomic Energy Research Institute (KAERI))
Suk Hoon KANG (Korea Atomic Energy Research Institute (KAERI))
Dong-Hyun AHN (Korea Atomic Energy Research Institute (KAERI))
Junhyun KWON (Korea Atomic Energy Research Institute (KAERI))
Sang-Yeob LIM (Korea Atomic Energy Research Institute (KAERI))
Seunghyun LEE (Korea Atomic Energy Research Institute (KAERI))
Dong Won LEE (Korea Atomic Energy Research Institute (KAERI))

## Topics

J. Materials Technology

## Communication type

Poster

## Abstract

Tungsten–CuCrZr divertor monoblock joints, developed for fusion reactor applications in Korea, must maintain high reliability under extreme thermal and irradiation conditions. This study investigates irradiation-induced changes in interfacial microstructure and mechanical properties through systematic characterization. Although the divertor monoblock is not a structural component, large thermal gradients and stresses during plasma operation and maintenance can promote cracking near the W/Cu interface, as previously predicted by finite element analyses.

Joint characterization using SEM, EBSD, EDS, and nanoindentation established baseline microstructural, compositional, and mechanical profiles across the tungsten, copper, and CuCrZr regions. Subsequent heavy-ion irradiation (9.3 MeV Fe at 320 °C) at the KAERI Heavy Ion Irradiation Facility enabled evaluation of irradiation effects on the W–Cu and Cu–CuCrZr interfaces through nano-hardness mapping and TEM analysis of FIB-prepared foils. The results reveal distinct irradiation hardening behaviors and defect microstructures near each interface, accompanied by elemental segregation that modifies interfacial stability. These findings provide key reference data for understanding degradation mechanisms in divertor monoblock joints under high-fluence irradiation and thermal loading, supporting the design of reliable plasma-facing components for advanced fusion devices.

# Microscale Tensile Behavior of Fe Ion-Irradiated Tungsten

Sun-Young PARK (Korea Atomic Energy Research Institute)
Suk-Hoon KANG (Korea Atomic Energy Research Institute)
Hyung-Ha JIN (Korea Atomic Energy Research Institute)
Junhyun KWON (Korea Atomic Energy Research Institute)
Dong-Hyun AHN (Korea Atomic Energy Research Institute)
Sangyeob LIM (Korea Atomic Energy Research Institute)
Seunghyun LEE (Korea Atomic Energy Research Institute)
Dong Won LEE (Korea Atomic Energy Research Institute)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

Tungsten has been extensively considered as a primary plasma-facing material for divertor components in future fusion reactors due to its high melting point, excellent thermal conductivity, and low sputtering yield. However, during fusion reactor operation, divertor materials are exposed to intense particle fluxes and irradiation environments that lead to significant irradiation damage accumulation. The formation and evolution of irradiation-induced defect structures can strongly influence the deformation behavior and mechanical performance of tungsten. Therefore, understanding the relationship between irradiation damage and the resulting mechanical response is essential for assessing the structural reliability of tungsten under fusion-relevant conditions.

In this study, the effect of ion irradiation on the mechanical properties of tungsten was investigated using Fe-ion irradiation. Irradiation experiments were conducted over a damage range of 1–20 displacements per atom (dpa) to represent different levels of irradiation damage at the KEARI Heavy Ion Irradiation Facility. Since ion irradiation typically produces a near-surface damaged layer with a limited penetration depth, microscale tensile specimens were fabricated to selectively probe the irradiated region. Dog-bone shaped micro-tensile specimens were prepared using focused ion beam (FIB) milling from within approximately 4 μm beneath the irradiated surface. Mechanical characterization was performed using in-situ micro-tensile testing inside a scanning electron microscope (SEM). This experimental approach enables direct observation of deformation processes during loading while simultaneously measuring the mechanical response of the irradiated material. In addition, microstructural characterization was carried out for each irradiation condition to investigate the evolution of irradiation-induced defects. The evolution of irradiation-induced defect structures was analyzed and correlated with the measured mechanical response, providing insight into the relationship between defect evolution and the deformation behavior of ion-irradiated tungsten.

Keywords: Divertor monoblock; Tungsten; Fe-ion irradiation; micro tensile testing;

# Waste management of tritiated materials: advances on tritium measurements capabilities to characterize physico-chemical mechanisms at stake

Elodie BERNARD (CEA IRFM)
Mickaël PAYET (CEA IRFM)
Olivier DEBELLEMANIERE (CEA SCBM)
Thibault CHARPENTIER (CEA NIMBE)
Carlos MORENO (CIEMAT - LABORATORIO NACIONAL DE FUSION)
Simon NIEMES ( Karlsruhe Institute of Technology (KIT))
Véronique MALARD (CEA BIAM)
Giorgio BAIOCCO (University of Pavia)
Sabina MARKELJ (Jozef Stefan Institute - JSI)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

Tritium ($^3$H), hydrogen radioactive isotope is a key radionuclide for future fusion devices, as it simultaneously represents a valuable fuel and a potential radiological hazard. Its intrinsic properties —high mobility in materials, volatility, and low radiotoxicity threshold— make its accurate measurement particularly challenging, yet essential for safety assessments, environmental monitoring and the management of future fusion nuclear facilities.

Horizon Europe project TITANS [1] has addressed cross-cutting activities required to improve knowledge $^3$H management at each step of the tritium life cycle, namely develop tritium release mitigation strategies, improve waste management and refine knowledge in the field of radiotoxicity, radiobiology and dosimetry. On most of these topics, the ability to rely on adequate in situ measurement of $^3$H activity is essential. However, developing reliable tritium metrology is complicated by the radionuclide's weak emissions, thus reducing direct detection, and by the radiological constraints that restrict the handling of highly active samples. Therefore, the project has addressed the coordinated development of comprehensive techniques able to probe $^3$H in surfaces, bulk materials and airborne particulates: autoradiography scintillation-based field detection ; nuclear reaction analysis ; laser-induced breakdown spectroscopy ; and Nuclear Magnetic Resonance (NMR), which will be the focus of this presentation.

NMR can analyse direct hydrogen isotope chemical interactions in materials, with unmatched sensitivity.using magic-angle sample-spinning NMR (MAS-NMR) [2]. The first 3H wide line NMR experiments were carried out on tritiated cementitious matrices incorporating getters (MnO2-Ag2O) envisioned for $^3$H waste storage, on small amount of particles (20 mg) with low tritium activity (20mBq/g). Spectrum for two types of cements particles (MKP and CEM-I) remained unchanged over 18 months, showing the stability of the trapped tritium. Continuous tritium degassing prevented MAS NMR measurements: this behaviour was linked with getter MnO2/Ag2O particles promoting a 3H/1H isotopic exchange with air humidity, as confirmed by 2H labelling strategy, and therefore severely jeopardizing its use as a tritium getter in sequestration matrices. The results confirm the capacity of NMR to probe hydrogen isotope content at the molecular scale, thereby offering a powerful characterization methodology for storage materials. The synergy with the other diagnostics developments provides complementary characterization of physico-chemical behavior, with the final goal being the development of field-deployable detection systems. The work contributes to strengthening the diagnostic toolbox required to assist safety, waste management, and fuel cycle optimization in present and future nuclear technologies.

[1] https://titans-project.eu/
[2] E. Nehlig & al. Nanoscale, 11 (2019) 8027–8036.

# Development of a Tritium Transport Prediction Code in Irradiation-Damaged Tungsten Considering Defect Accumulation and Multiple Hydrogen Trapping

Makoto KOBAYASHI (National Institute for Fusion Science)
Teruya TANAKA (Natinal Institute for Fusion Science)
Yosuke HARASHIMA (Nara Institute of Science and Technology)

## Topics

J. Materials Technology

## Communication type

Oral

## Abstract

In fusion reactors, tungsten will be exposed to intense neutron irradiation, high heat flux, helium ash, and tritium-containing fuel plasma. Neutron irradiation generates various defects in tungsten, and the migration and interaction of these defects lead to the formation and evolution of defect clusters. These irradiation-induced defects act as strong trapping sites for hydrogen isotopes and play a critical role in determining the tritium inventory in plasma-facing components.

Recent studies have reported that vacancy clusters, one of the dominant irradiation defects, can trap multiple hydrogen isotope atoms. Moreover, the trapping energy and thermal stability of hydrogen isotopes in vacancy clusters depend on the number of coexisting hydrogen atoms in the cluster. However, most existing tritium transport models treat irradiation defects as static traps with fixed trapping capacities, and the dynamic evolution of irradiation defects and their multiple trapping capacities have not been sufficiently incorporated.

In this study, we developed a tritium transport prediction code that couples a hydrogen isotope migration model with a rate-theory-based model describing the accumulation and annealing of irradiation-induced defect clusters. The hydrogen isotope transport model includes diffusion, trapping, and detrapping processes at irradiation defects, while the defect evolution model describes the nucleation, growth, and annihilation of defect clusters under irradiation. In addition, the model explicitly incorporates the multiple hydrogen trapping capacity of vacancy clusters.

Using the developed code, we simulate hydrogen isotope retention and release behavior in irradiation-damaged tungsten and evaluate the impact of defect evolution and multiple trapping effects on tritium inventory. The model predictions are compared with experimental results obtained for damaged tungsten. Furthermore, recent advances in experiments and modeling related to hydrogen isotope trapping in irradiation defects will be discussed.

#14029

# An overview of UKAEA activities in the development and implementation of EU-DEMO design criteria for in-vessel components

Wang YIQIANG (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Giacomo AIELLO (EUROfusion PMU, Boltzmannstrasse 2, 85748 Garching, Germany)
Sreeram MADABUSI (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Vishnu AMBIKADEVI (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Sebastian ROSINI (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Samad KHANI (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Zsolt VIZVARY (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Peter COOPER (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Neelam GUPTA (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Bernard LIEW (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Jacob TAYLOR (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Aleksandr ZINOVEV (Belgian Nuclear Research Centre, SCK CEN, Mol, 2400, Belgium)
Dmitry TERENTYEV (Belgian Nuclear Research Centre, SCK CEN, Mol, 2400, Belgium)
Cory HAMELIN (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Thomas BARRETT (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Gerald PINTSUK (Forschungszentrum Jülich GmbH, Institute of Fusion Energy and Nuclear Waste Management – Plasma Physics, 52425 Jülich, Germany)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

After decades of worldwide research and development including recent world records for inertial and magnetic fusion energy, fusion engineering and technology is progressing steadily from experiment and concepts towards prototype and commercial fusion power plants. Over the next few years, focus will be shifting from plasma physics to addressing engineering and technology realisation challenges, from conceptual design to engineering design and licensing. A critical step to enable this transition is to establish fusion design criteria covering intended use in light of emerging concepts, in particularly for the in-vessel components.

In this work, we provide an overview of UKAEA's contributions to the development of European DEMO design criteria for in-vessel components (DDC-IC) which is a collaborative international project within EUROfusion's Work Package Materials, aimed at establishing advanced inelastic design rules and design-by-analysis methods that address shortcomings within existing design codes and standards. These rules encompass both deterministic and probabilistic approaches and are essential for verifying designs and predicting the lifespan of fusion systems, structures, and components.

Furthermore, the implementation of the DDC-IC for both WPDIV and WPBB is investigated, ensuring alignment with the needs of component design, assessment and the associated verification and validation processes. The remaining technical challenges are identified, and a roadmap is proposed within EUROfusion to address them. By integrating these efforts, we aim to establish a comprehensive pathway to address engineering challenges, develop fusion specific design criteria, and ensure compliance with regulatory requirements, ultimately advancing the path toward commercial fusion energy.

#14043

# Benchmarking neutron transport and shielding performance in fusion-relevant concrete using VERDI activation detectors at ENEA FNG

Theodora VASILOPOULOU (INRASTES, National Centre for Scientific Research Demokritos, Neapoleos 27, 15310, Athens, Greece)
Davide FLAMMINI (ENEA, Nuclear Department (NUC-FUSEN), Frascati, via E. Fermi 45, 00044 Frascati (RM), Italy)
Stefano LORETI (ENEA, Nuclear Department (NUC-FUSEN), Frascati, via E. Fermi 45, 00044 Frascati (RM), Italy)
Andrea COLANGELI (ENEA, Nuclear Department (NUC-FUSEN), Frascati, via E. Fermi 45, 00044 Frascati (RM), Italy)
Roberto LICCIONE (Department of Industrial Engineering, University of Rome “Tor Vergata”, Via del Politecnico 1, 00133, Rome)
Virginie LOMBARDI (Sapienza University of Rome, Department of Electrical and Energy Engineering, Corso Vittorio Emanuele II 244, 00186, Rome, Italy)
Ion-Evangelos STAMATELATOS (INRASTES, National Centre for Scientific Research Demokritos, Neapoleos 27, 15310, Athens, Greece)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

A benchmark experiment was performed at the ENEA Frascati Neutron Generator (FNG) to validate neutron transport calculations and assess the impact of nuclear data and material composition on the shielding performance of concrete intended for DEMO and IFMIF/DONES fusion facilities. Two VERDI detectors were fabricated and positioned at selected locations along the ENEA FNG concrete shielding mock-up to measure neutron fluence and determine the neutron energy spectrum under 14 MeV neutron irradiation. The VERDI detectors consist of a set of well-characterized metallic elements encapsulated in low-activation capsules. They were developed to enable accurate determination of neutron fluence and energy spectra while withstanding the harsh environmental conditions expected in fusion devices and power plants. After irradiation, the detectors were analyzed using gamma-ray spectrometry to determine the activity induced in each metallic element contained within the capsules. The measured activities of the activated metallic elements were combined with neutron cross-section data for the corresponding nuclear reactions to generate the response functions required for neutron spectrum unfolding. The unfolding procedure was performed using the maximum entropy method.
This work presents the results obtained from the VERDI measurements in the FNG concrete benchmark experiment and compares them with simulations carried out using a detailed MCNP model of the experimental set-up coupled with the FISPACT-II radionuclide inventory code. The results provide valuable experimental data for benchmarking neutron transport calculations in the ENEA FNG concrete assembly and demonstrate the feasibility of using VERDI detectors to determine neutron fluence in shielding components of future fusion reactors.

This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 — EUROfusion). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.

# Assessment of fusion-relevant ductility exhaustion criteria for CuCrZr

Peter COOPER (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Matthew CARRINGTON (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Yiqiang WANG (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)
Alex MARSH (UKAEA (United Kingdom Atomic Energy Authority), Culham Campus, Abingdon, Oxfordshire OX14 3DB, United Kingdom)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

Copper Chromium Zirconium (CuCrZr) alloys are candidate materials for use in fusion applications where high heat flux loading is expected, given their superior thermal and mechanical performance relative to pure copper. Off-normal high heat fluxes and mechanical loads caused by plasma instability are expected in the operation of first of a kind fusion machines. Understanding plastic failure in this material is essential when predicting the consequences of damage to these components under such loading. With good plastic failure predictions, components can be designed so that, even when damaged, their pressure boundary will not be compromised thus preventing loss of coolant events.

In this paper two ductility exhaustion criteria from reputable codes - the ASME Boiler Pressure Vessel Code and the ITER Structural Design Code for In-vessel Components - are compared against experimental tests of CuCrZr samples to failure. The comparison is made over a range of stress states. Discrepancies between the test data and design limits from each criterion are quantified.

Reasons for the observed discrepancies are given by considering things such as Lode parameter, path dependency, and the nonlinear relation between plastic strain and void volume fraction. This assessment provides a roadmap for the improvement of plastic failure criteria that are particularly necessary when designing CuCrZr-based fusion components, given the stress states occurring during off-normal events have been found in the design space where some of the largest discrepancies are observed in the present study.

This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 — EUROfusion) and from the EPSRC [grant number EP/W006839/1]. Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.

# Irradiation of bulk fusion materials with 10-30 MeV protons at MIT's Schmidt Laboratory for Materials in Nuclear Technologies

Zachary HARTWIG (Massachusetts Institute of Technology)
Louis WAINWRIGHT (Massachusetts Institute of Technology)
Sarah CHAMBERLAIN (Massachusetts Institute of Technology)
Andre CHOUINARD (Massachusetts Institute of Technology)
Jose ESTRADA (Massachusetts Institute of Technology)
Sara FERRY (Massachusetts Institute of Technology)
James RIDZON (Massachusetts Institute of Technology)
Rui VIEIRA (Massachusetts Institute of Technology)
Amy WATTERSON (Massachusetts Institute of Technology)
Rick LECCACORVI (Massachusetts Institute of Technology)
Emily NEILL (Massachusetts Institute of Technology)
Evan LAMBERT (Massachusetts Institute of Technology)
Adam MULLIN (Massachusetts Institute of Technology)
Trevor MARCHHART (Massachusetts Institute of Technology)
Julian ROCHER (Massachusetts Institute of Technology)
Matthew BALDWIN (University of California, San Diego)
Sophie BLONDEL (University of Tennessee, Knoxville)
Adrien COUET (University of Madison, Wisconsin)
David SPROUSTER (Stony Brook University)
Lianyi CHEN (University of Madison, Wisconsin)

### Topics

J. Materials Technology

### Communication type

Oral

### Abstract

Materials in deuterium-tritium fusion power plants will experience up to ~15 displacements per atom (dpa) per year and 150 atomic parts per million per year of transmutation-induced helium. The uncertainties around how such changes will impact the structural, thermal, and electrical property evolution of fusion materials must be minimized to provide confidence in the design and operation of first-of-a-kind systems being proposed for the 2030s. To meet this goal, the direct irradiation of bulk material specimens by 10-30 MeV protons has recently been explored at MIT for rapid, high fidelity irradiation and testing of fusion materials. Advantages include high damage rates (~1 dpa/day), fusion-relevant helium transmutation (up to ~10 He_appm/dpa), and the use of macroscopic (>200 um) specimens that enable extraction of engineering properties.

The Schmidt Laboratory for Materials in Nuclear Technologies (LMNT) at the MIT Plasma Science and Fusion Center will utilize this technique in a purpose-built facility for 10-30 MeV proton irradiation and testing of bulk materials. Slated to begin operation in summer 2026, LMNT will employ a 30 MeV, 800 uA, dual-extraction proton cyclotron with external beam lines connecting to four independent and modular end stations for materials irradiation. The end stations are capable of high beam power handling and can be configured to impose a wide range of environmental conditions on materials during irradiation, such as high temperature or liquid coolant. Subscale irradiation specimens are adopted from the experiences of neutron irradiation in the fission and fusion communities, where minimizing the testing volumes are advantageous to maximize throughput, flux uniformity, and post-irradiation safety while preserving bulk property measurement capability.

The objective of this talk is to introduce LMNT to the fusion materials community. The talk will review the proposed advantages of 10-30 MeV proton irradiation and then cover the present status of LMNT and preliminary results from cyclotron commissioning and the first bulk material irradiations. The talk will also briefly cover the work of the FLAME Collaborative, a five institution U.S. collaboration funded by the U.S. Department of Energy to utilize LMNT to advance experimental and computational fusion materials science. Importantly, the talk will emphasize that LMNT should be seen as highly synergistic with other efforts to advance our understanding of materials in fusion environments, such as prototypic neutron source efforts in the U.S. (e.g. FPNS) and internationally (e.g. IFMIF).