Materials and Innovations for Fusion Energy
Room 321 - Session FUS-TuA

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Moderators: Dr. Eric Dombrowski, Commonwealth Fusion Systems, **Dr. Zachary Robinson**, Laboratory for Laser Energetics, The University of Rochester

4:00pm **FUS-TuA-8 Challenges in Tritium Surface Characterisation: From Memory Effects to Fusion byproducts management**, ***Florian Priester***, *Dominic Batzler, Robin Größle, Michael Sturm, Simon Niemes, Alexander Marsteller,* Karlsruhe Institute of Technology - Tritium Laboratory, Germany **INVITED**

The characterisation of tritium (T_2) in materials presents a set of unique challenges that distinguish it significantly from its stable isotopes, hydrogen (H_2) and deuterium (D_2). In vacuum environments and analytical devices, tritium's radioactive nature and high mobility lead to a so-called "memory effect"—the adsorption and diffusion of tritium into system surfaces—which complicates maintenance protocols, degrades analytical precision, and artificially raises the Limit of Detection (LoD).

This presentation evaluates the current suite of measurement technologies utilised for tritium quantification, including calorimetry, Beta-Induced X-ray Spectroscopy (BIXS) for (near-) surface analysis, and liquid scintillation counting. A critical bottleneck in the field remains the sound determination of decontamination factors across these methodologies.

Furthermore, we explore some examples to assess material properties with regard to tritium sticking and decontamination possibilities. Beyond the technical hurdles, there is a burgeoning economic demand for "cheap" and accessible detection methods. In the absence of reliable, cost-effective quantification, facilities are forced into the conservative—and significantly more expensive—declaration of byproducts as intermediate-level waste. This work underscores the necessity of advancing surface characterisation techniques to enable the economic declassification of materials and streamline future fusion and nuclear waste management.

4:30pm **FUS-TuA-10 Evaluating the Performance of Next Generation Plasma-Facing Materials**, ***Robert Kolasinski,*** Sandia National Laboratories*; Kent Christian,* Stony Brook University*; Jonathan Coburn, Antonio Cruz, Feng-Jen Chang, Mary Alice Cusentino,* Sandia National Laboratories*; Lance Snead, Jason Trelewicz, David Sprouster,* Stony Brook University **INVITED**

The science of plasma-material interactions (PMI) is fundamental to the realization of magnetic fusion as an energy source, and predicting how materials behave in the extreme environments of fusion devices remains one of the most daunting challenges in materials science. Surfaces directly exposed to intense plasmas will be continually reconfigured over their operational lifetime. This evolving surface structure governs material degradation and is also intimately coupled to neutral and impurity recycling and edge turbulence in the core plasma.

This presentation will provide an overview of recent laboratory testing and surface characterization of advanced tungsten alloys and ceramics that are among the leading candidate plasma-facing materials for fusion devices. The work combines experiments and modeling to better understand the effects of high-flux plasma exposure on these materials. Although they offer promising thermomechanical properties and microstructural stability, important gaps remain in our understanding of their response, including erosion/redeposition, near-surface defect nucleation and growth, and tritium uptake. Recent efforts have focused on evaluating these materials in a high-flux steady-state plasma source, coupled with in situ and post-mortem diagnostics to track the evolution of surface composition and structure. Initial studies of doped and dispersoid-strengthened tungsten have yielded promising results, including improved resistance to recrystallization and plasma-induced surface morphology changes. Likewise, ultra-high-temperature ceramics are emerging as attractive alternatives to tungsten, with the potential for greater resilience to neutron damage. These laboratory studies have motivated recent experiments at tokamak facilities, where tungsten and ceramic microstructures are exposed to reactor-relevant heat and particle fluxes. Such experiments enable systematic evaluation of how compositional and microstructural variations improve surface resilience under combined loading. Together with complementary modeling across multiple length scales, these results help identify promising pathways for further materials optimization.

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5:00pm **FUS-TuA-12 Optimised Variable-Temperature Cryocooler Design for Reduced Tritium Inventory and Operational Risk**, *Pankaj Sharma, Robert Buchanan, Adam Bruce,* ***William Schofield,*** UKAEA, UK

Thermal conducting materials (TCMs) are commonly used at cryogenic interfaces to improve heat transfer between a cryocooler cold head and a sample cell. However, in tritium-handling systems, conventional TCMs may introduce hydrogen-containing materials that contribute to isotope exchange, tritium retention, and increased operational complexity. This study evaluates whether TCMs are essential for achieving the required thermal performance of a cryogenic adsorption system, or whether their function can be replaced through optimised mechanical interface design. Controlled heat transfer experiments were performed at four target temperatures (75, 150, 200, and 323 K), representing the operating range of planned adsorption studies. Baseline experiments were conducted using the original cold head-sample cell assembly, both with and without a Type 120 silicone TCM interface. Cold head and sample cell temperatures were monitored under steady-state conditions to assess thermal coupling efficiency. Results showed that the TCM-assisted configuration provided effective thermal coupling, with sample cell temperatures closely matching cold head temperatures across the full operating range. The temperature deviation between the cold head and sample cell remained minimal, with variations of approximately 0.3 K at 150, 200, and 323 K, and ~0.75 K at 75 K. In contrast, removal of the TCM introduced significant interfacial thermal resistance, resulting in substantially larger temperature deviations (up to ~38.2 K) and slower thermal response, particularly under cryogenic conditions at 75 K. To eliminate TCM usage, alternative mechanical interface designs were developed and tested. Through optimisation, a modified assembly achieved the required improved thermal performance without the use of TCMs.

5:15pm **FUS-TuA-13 Pellet Flight Path Design Challenges and Solutions for ITER**, ***Matt Williamson***, *Sarah Smith, Steven Meitner, Eileen Sibley, Oscar Martinez,* Oak Ridge National Laboratory

The ITER Fuelling Pellet Injection System (FPIS) will provide hydrogenic, cylindrical pellets for steady-state plasma core fueling and control of Edge Localized Mode instabilities. ITER will be the world's largest tokamak, a magnetic confinement fusion device with the primary mission to demonstrate the scientific and technological feasibility of fusion energy. In the final ITER machine configuration FPIS injectors will be housed in three secondary tritium containment casks, with one cask allocated to each of three lower-level ports positioned 120° apart. For each cask, there will be three pellet flight lines with penetrations through the cryostat and vacuum vessel (VV). Pellets will be extruded, sized, propelled through the flight line selector, and guided to the plasma injection points located in the VV via flight lines.

The pellet flight path for ITER presents a variety of design challenges. Pellets are required to survive gap jumps through isolation valves, propellant removal pumping stages, diagnostic cavities, and mechanical connections required to permit phased installation of the FPIS. Pellet-facing flight path components adjacent to these gap jumps have been designed to minimize impact on pellet survivability while allowing for manufacturability, assembly, and removal. Portions of the flight path must also accommodate radial displacement of the VV during machine operation while maintaining fuel confinement and containment. Known displacements and as-built misalignments must be absorbed by the design while maintaining a robust pellet flight path and secondary vacuum containment. The pellet flight path must endure high temperatures, electromagnetic displacements, and pass through several interfacing ITER subsystems.

The ITER pellet flight path design must provide reliable pellet delivery while allowing flexibility for machine as-built installation, complying with material requirements for plasma purity and high-vacuum environments, and withstanding harsh environmental conditions for decades of machine operation. These key design challenges must be resolved to successfully fuel ITER and inform practical fueling systems deployed at future fusion plants.

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5:30pm **FUS-TuA-14 Westinghouse'S Experience in the Fabrication and Welding of ITER'S Vacuum Vessel Sectors**, *Stefano Batticci,* Westinghouse Italy; *Simon Roche,* Westinghouse Electric France; ***Paolo Ferroni,*** Westinghouse Electric Company

Westinghouse is most often known as a world leader in the development of nuclear fission power plants, including manufacturing of nuclear fuel and major equipment required to best exploit energy from nuclear fission. However, for several years Westinghouse has not only been following advances in fusion energy – while being agnostic to the type of fusion technology being pursued – but has also been involved in fusion technology development as Service Provider. This role has found its most notable example in the manufacturing, together with Ansaldo Nucleare and Walter Tosto (collectively known as the AMW Consortium), of five out of the nine sectors comprising the ITER's Vacuum Vessel (VV) (see Figures 1 and 2), and in the welding of all nine sectors which is an operation currently in qualification phase. In this presentation, Westinghouse will describe the manufacturing strategy utilized for the ITER's VV sectors, including:

1. Review of the Conceptual Design received from the ITER Organization and suggesting several changes to improve manufacturability;
2. Development of the Detailed Manufacturing Design (fabrication models and drawings) and manufacturing engineering (i.e., assembly sequence, , work sequences, development of dedicated jigs and tools needed for fabrication, re-verse engineering for CAM Computer Aided Machining purposes, NDE inspection plans, etc.) for the items to be fabricated;
3. Development of the manufacturing engineering for the final assembly phase of all European Sectors, including the development of the Virtual Fitting process as main engineering tool for the assembly of extra large components.

In addition, information will be provided on the preparatory activities for the in-situ welding of all nine VV sectors. We will describe what Westinghouse proposed to the ITER Organization to weld the 9 VV Sectors in parallel at the same time. Accounting for the pre-qualification phase we anticipate a 30 month welding sequence on-site inside the ITER Tokamak Pit:

1. Phase 1-2-3 – Feasibility Study
2. Phase 4 – Pre-qualification: Trials Mechanical, machining, reverse engineering, welding, NDT, RT Linac, logistics
3. Phase 5 – Vacuum Vessel Sectors Welding. Welding preparation: qualification process, resource training, tooling procurement, anticipated operations. Welding phase: welding/machining operation, post welding NDE

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