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

Materials and Innovations for Fusion Energy

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.

SNL is managed and operated by NTESS under DOE NNSA contract DE-NA0003525

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.

This work was supported by the U.S. Department of Energy contract DE-AC05-00OR22725. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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

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

Fusion Materials, Plasma-Facing Components, and Tritium Engineering

Moderators: Dr. Dale Hitchcock, Savannah River National Lab, **Dr. Matthew Sharpe**, The Laboratory for Laser Energetics, The University of Rochester

8:00am **FUS-WeM-1 Water Detritiation: From ML-Guided Selection to Catalyst Synthesis*, Rashad Ahmadov,*** *Zachary Robinson, Mark Wittman, Matthew Sharpe,* Laboratory for Laser Energetics

Building on prior validation of a hydrogen-based surrogate unit for water detritiation, this workadvances catalyst development through integrated computational screening and targeted synthesis. The previous study established baseline performance using commercial Cu-Zn catalysts, revealing reaction rates limited by the internal mass diffusion. To identify candidates with potentially improved characteristics, Machine Learning (ML)-accelerated adsorption energy calculations (AdsorbML) were employed to screen a broader alloy design space. A promising composition was selected based on predicted surface energetics favorable for water splitting reaction adsorbates. The candidate catalyst was synthesized and characterized using XRD and XPS to confirm phase composition and surface chemistry. Temperature-programmed reduction (TPR) was performed to determine appropriate activation conditions prior to reactivity testing. The catalyst was then tested in the surrogate reactor under conditions comparable to prior experiments. Preliminary results confirm catalytic activity for hydrogen evolution with oxygen retention. This work demonstrates a computationally-guided pathway for catalyst discovery for thermochemical water detritiation, contributing toward optimized tritium recovery systems for closing the fusion fuel cycle.

This material is based upon work supported by the Department of Energy [National Nuclear Security Administration] University of Rochester "National
Inertial Confinement Fusion Program" under Award Number(s) DE-NA0004144.

8:15am **FUS-WeM-2 Al Intercalation Into C-Vacancy Sites in TiCx for Low-Temperature Synthesis of MAX-Phase Ti_2AlC Hydrogen Permeation Barrier Coatings*, Isabella Stepanek,*** Georgia Institute of Technology*; Dale Hitchcock,* Savannah River National Lab*; Eric Vogel,* Georgia Institute of Technology

MAX-phase Ti_2AlC thin films were synthesized using PVD magnetron sputtering of a bilayer system that utilizes an Al precursor layer with a subsequent TiAlC layer, followed by vacuum annealing at 650°C for 3h up to 20h. Ex situ XRD of TiAlC only films (no Al precursor layer) reveal that annealing drives Al out of the $Ti_{0.66}Al_{0.34}C$ rocksalt phase present in the as-deposited films, generating additional C-vacancy sites to form $TiC_{0.5}$. No Ti_2AlC was synthesized from these TiAlC only films. With the utilization of an Al precursor layer deposited prior to the TiAlC layer, it was found that the as-deposited bilayer contains spatially distinct $TiC_{0.56}$ and Al phases, establishing the diffusion geometry required for Al intercalation-driven Ti_2AlC formation. This mechanism is initiated via vacuum annealing at 650°C which mobilizes Al both from the $Ti_{0.66}Al_{0.34}C$ phase in the TiAlC layer but also Al from the precursor layer. The resulting upward Al diffusion and intercalation into ordered C-vacancy planes of $TiC_{0.55}$ is identified as the primary mechanism driving low-temperature Ti_2AlC formation in Al/TiAlC films. XRD results also show the formation of Ti_3Al along with the $TiC_{0.55}$ and Ti_2AlC phases, mirroring reported two-step reaction pathways that include the formation of Ti-Al intermetallics and TiC_x that react to form Ti_2AlC. Ex situ XPS depth profiles corroborate this mechanism, showing progressive Al depletion from the precursor layer into the TiAlC layer with annealing time. Associated scans of the Al 2p orbital taken during depth profiles shows binding energy shifts evidencing changes in the Al chemical environment consistent with C-vacancy incorporation and Ti_2AlC synthesis.

These efforts towards low-temperature synthesis of MAX-phase coatings allow for their potential use on low activation structural materials used in fusion reactor blanket components. Ti_2AlC is a particularly promising candidate for this application, as aluminum-containing MAX-phases form a dense, protective Al_2O_3 scale upon oxidation that has been demonstrated to reduce hydrogen isotope permeation through structural materials.

8:30am **FUS-WeM-3 Characterizing Heavy Water Displacement in Molecular Sieve Drying Beds for Practical Trace Tritium Capture Methods*, Brandon Massett,*** *Walter Shmayda,* Tritium Solutions Inc.

Tritium plays a crucial role in nuclear fusion power plant designs, and adsorption beds are essential tools for managing tritiated water vapor. A series of tests was performed to investigate whether a saturated adsorption bed preferentially adsorbs heavy water vapor. The design of passive tritium control systems could potentially rely on adsorption beds preferentially trapping heavier isotopologues of water. This work investigates the displacement phenomenon, as well as the effects of carrier gas superficial velocity and heavy water concentration on bed performance. Significant displacement was observed when a humid stream containing heavy water was diverted through a bed pre-saturated with light water, as indicated by changes in the partial pressures of D2O and H2O. Following the capture of heavy water in the bed, the subsequent rise in D2O partial pressure depended on both the superficial gas velocity and the heavy water humidity in the gas stream. Higher superficial velocities and humidities led to faster and steeper mass transfer profiles within the adsorption bed. These heavy water breakthrough curves were fit and compared with theoretical models for adsorption beds operating under standard conditions.

8:45am **FUS-WeM-4 Characterizing the Behavior of Hydrogen Isotopes and Impurity Species on Vanadium Surfaces by Direct Recoil Spectroscopy*, Feng-Jen Chang,*** *Antonio Cruz, Mary Alice Cusentino, Robert Kolasinski,* Sandia National Laboratories

Due to their high permeability for hydrogen isotopes (HI), vanadium (V) membranes are being considered for separation of HI from the plasma exhaust of future fusion reactors. Surface effects, particularly the presence of impurities, have a large influence on the performance of such membranes. However, it is often difficult for conventional surface analysis tools to directly observe how HI interacts with other surface species. Ion scattering spectroscopy (ISS) and direct recoil spectroscopy (DRS) offer the advantage of being able to directly detect adsorbed HI by analyzing the energies of scattered and recoiled particles from ion scattering on the surface.

In this study, we use a 2 keV Ne^+ ion beam to probe the surface composition of a polycrystalline V sample while dosing with hydrogen (H) or deuterium (D) with increasing pressure [Fig. S1(a)]. Gas species and surface temperatures (T_s) are varied for comparison, and a gas doser heated to 1100 °C is available for dissociating H_2/D_2 molecules into H/D atoms. DRS peaks of H/D are clearly identified in the ion energy spectra [Fig. S1(b)]. Pressure dependences of the peak intensities can be well fitted by a simple site-blocking model with Langmuir isotherm [Fig. S1(c)], which assumes that adsorbates only interact with each other by competing for empty adsorption sites. Such fittings reveal the change of the adsorption-desorption balance of HI with different surface conditions. The ratio between the adsorption rate coefficient and desorption rate coefficient, $K = k_{ad} / k_{de}$, decreases with increasing T_s, indicating that desorption is stronger at higher T_s [Fig. S1(d)]. K is also higher with atomic H/D dosing than with H_2/D_2 dosing, since atomic species have higher chemical energy to overcome the adsorption energy barrier, and thus result in higher adsorption rates.

In addition, preliminary studies of oxygen (O) adsorption by DRS while dosing the V surface with O_2 have been performed. The DRS O peak intensity growth with increasing O_2 pressure is similar to that measured by X-ray photoelectron spectroscopy (XPS) [Fig. S1 (e)], which validates the DRS results. The ongoing work includes sequentially dosing O_2 and HI to validate the site-blocking model. Finally, investigation of the effect of permeation barrier coatings, such as SiC, W or Al_2O_3, on V and V-alloys with both DRS measurements and density functional theory (DFT) modelling is also underway, and preliminary results from this work will be included in this presentation.

SNL is managed and operated by NTESS under DOE NNSA contract DE-NA0003525

9:00am **FUS-WeM-5 Theoretical and Experimental Study of Hydrogen Super-Permeation in Iron*, Masashi Shimada,*** *Thomas F. Fuerst,* Idaho National Laboratory **INVITED**

Metal foil pumps (MFPs) exploit plasma–driven hydrogen super–permeation to selectively separate and compress hydrogen isotopes (D/T) from helium and impurity species in the fusion exhaust stream, forming a key enabling technology for Direct Internal Recycling (DIR)–based fuel–cycle

architectures. Despite their promise, the technical readiness level of MFPs remains insufficient for integration into next–generation fusion systems. Advancing the understanding of hydrogen super–permeation in metallic membranes is therefore essential for developing high–performance pumping components capable of operating under reactor–relevant plasma conditions.

This work presents an integrated experimental and theoretical investigation of deuterium super–permeation in high–purity α–Fe. Plasma–driven permeation experiments conducted in the Tritium Plasma Experiment (TPE) at Idaho National Laboratory demonstrated that a 99.99% α–Fe foil can achieve a maximum pumping throughput of 3.0 L s^{-1} cm^{-2} and a pumping speed of 3.0 m^3 s^{-1} m^{-2} at 523 K under a D_2 pressure of 0.013 Pa. The permeation behavior was found to be strongly governed by the surface sticking coefficient, α, and by the incident D^+ ion energy. Comparison with the Pick–Sonnenberg model showed excellent agreement when using a previously measured $\alpha \approx 1.6 \times 10^{-9}$, with peak pumping performance occurring near an incident ion energy of 10 eV [1]. In contrast, incident energies above 50 eV caused a sharp decline in pumping performance, consistent with sputtering–induced removal of the native oxide monolayer—an effect also reported in prior studies of Nb and Fe.

A complementary theoretical study applied the Pick–Sonnenberg formulation to evaluate hydrogen permeation in candidate group–V membrane materials, focusing on vanadium with an experimentally measured effective permeability influenced by surface oxide and subsurface carbide layers [1]. Modeling showed that even with the reduced permeability, an uncoated 1 mm–thick V membrane can achieve approximately 10% hydrogen permeation at 973 K when operating with $\alpha \approx 10^{-3}$ and ion fluxes near 10^{20} m^{-2} s^{-1}. These findings highlight the potential for high–temperature operation to enable attractive pumping performance without the Pd coatings typically required for group–V metals.

Collectively, these results deepen the mechanistic understanding of hydrogen super–permeation in metallic membranes and provide a physics–grounded basis for optimizing high–flux MFP systems within advanced fusion fuel–cycle concepts.

Reference:

[1]M.A. Pick and K. Sonnenberg, *J. Nucl. Mater.* **131** (1985) 208-220

9:30am **FUS-WeM-7 Development of Tungsten Alloys to Mitigate Impacts of Neutron Irradition on Tritium Inventory***, **Yuji Hatano**, Qicong Chen, Yuga Kimura, Naoko Oono, Katsuya Suzuki,* Tohoku University, Japan*; Tatsuya Kuwabara,* Aichi Institute of Technology, Japan*; Koji Inoue,* Tohoku University, Japan*; Jing Wang,* Hefei University of Technology, China **INVITED**

While Tungsten (W) is a leading candidate for a plasma-facing material in fusion reactors, neutron-induced displacement damage creates vacancy-type defects that significantly increase tritium retention [1,2]. To mitigate this, we developed W alloys using undersized solute elements like chromium (Cr) and rhenium (Re). These elements promote the annihilation of vacancy-type defects during irradiation at temperatures ≥773 K and reduce the resulting trap density [3-8]. Furthermore, recent findings show these alloying elements enhance defect recovery during post-irradiation annealing at approximately 1473 K, offering a potential pathway for in-vessel trap removal during scheduled maintenance. This presentation explores the mechanisms driving these enhanced annihilation and recovery processes.

[1]Y. Hatano et al., Nucl. Fusion, 53 (2013) 073006.

[2]Y. Hatano et al., J. Nucl. Mater., 438 (2013) S114.

[3]Y. Hatano et al., Nucl. Mater. Energy, 9 (2016) 93.

[4]J. Wang et al., J. Nucl. Mater., 545 (2021) 152749.

[5]J. Wang et al., J. Nucl. Mater., 559 (2022) 153449.

[6]J. Wang et al., Mater. Design, 229 (2023) 111899.

[7]J. Wang et al., Mater. Design, 259 (2025) 114783.

[8]Y. Nobuta et al., J. Nucl. Mater., 566 (2022) 153774.

11:00am **FUS-WeM-13 Liquid Metals as Plasma-Facing Components in Fusion Devices: Can They Solve the Power Handling Challenge?** *, **Martin Nieto-Perez**, Calixto Alvarado,* Pennsylvania State University*; Daniel Andruczyk,* University of Illinois at Urbana-Champaign*; Lane Carasik,* Virginia Commonwealth University*; Ama Dahanayakle,* Pennsylvania State University*; Bruce Koel,* Princeton University*; Mike Kotschenreuther,* Exofusion*; Rajesh Maingi,* Princeton Plasma Physics Laboratory*; David Ruzic,* University of Illinois at Urbana-Champaign*; Sergei Smolentsev,* Oak Ridge National Laboratory*; Vlad Soukhanovskii,* Lawrence Livermore National Laboratory*; Xing Wang,* Pennsylvania State University*; Kevin Woller,* Massachusetts Institute of Technology

One of the most daunting challenges on the road to commercial fusion machines is the development of materials and components capable of withstanding the enormous power loads associated with particle leakage from magnetically confined plasmas; in addition to being resilient enough to withstand very high heat loads, these components need to facilitate the recovery of any valuable tritium fuel that reaches them. To satisfy these requirements, plasma-facing component systems based on liquid metals are considered a viable option, but remain at a lower technology readiness level (TRL) than their solid counterparts. To advance the maturity of these technologies, a DOE-funded initiative, Liquid Metals: Advancing the technical Readiness of plasma-facing Materials Operational Response (LM-ARMOR), has been established, spanning multiple federal laboratories and academic institutions. The overarching aim uses a combination of dedicated experiments and targeted calculations to explore and develop different PFC systems based on liquid metals. This collaboration aims to derisk LM technologies by addressing specific scientific and technological gaps that must be overcome to develop next-step integrated fusion test facilities and fusion reactors. The four overarching topics are material compatibility assessments, MHD flows, development of novel liquid metal alloys, and integrated testing. In this talk, efforts to develop the capillary pore system and to study novel metal alloys will be highlighted, emphasizing the need for experimental measurements of critical properties, including wetting, hydraulic permeability, sputtering, and surface segregation.

11:15am **FUS-WeM-14 Atomic Layer Deposition Affecting Gas Retention in Fusion Targets***, **Mark Bonino**,* University of Rochester

Glow discharge polymer (GDP) is the primary material used as an ablator for spherical, direct-drive laser experiments at the University of Rochester's Laboratory for Laser Energetics. GDP capsules are millimeter-scale hollow spheres with gas permeation time constants of ~1 min for hydrogen isotopes—the gases used for fusion reactions. Gas permeation time constants, however, must be significantly longer (tens of hours) to meet the specified fuel pressure at the time of the laser shot. Most experiments require fuel pressures of 15 atm±10%. Sputter-deposited aluminum has been demonstrated to increase the gas permeation time constant, using layers of around 100 nm. However, sputtered aluminum films often exhibit a broad distribution of time constants, rendering many samples unusable. An alternative to sputter deposition is atomic layer deposition (ALD), where layers of Al_2O_3 are applied using a vapor-phase deposition process to produce a contiguous gas-permeation layer. ALD is an alternative method to increase hydrogen gas retention and is investigated in this study.

The presented work will report gas-permeation time constants on spherical and cylindrical plastic shells from a decay curve measured using a residual gas analyzer. In our time-constant measurement system, the targets are pumped out to high vacuum (10^{-6} Torr) and filled to pressures up to 1000 Torr. Preliminary observations show time constants ranging from 1000s to 2000s, with a dependence on pressure. Changes in the permeation rate as a function of cycling will be investigated. Characterization (using x-ray photoelectron spectroscopy, a scanning electron microscope, and an atomic force microscope) will be performed to observe microscopic changes on the sample surfaces that impact gas retention.

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Presentation Type: Oral presentation

Presenter: Mark Bonino, mbon@lle.rochester.edu [mailto:mbon@lle.rochester.edu], (585)275-1404 (office)

11:30am **FUS-WeM-15 Measurement of Residual Impurities after a Cryogenic DT Inertial Confinement Fusion (ICF) Implosion***, **Errol Alden**, Zak Robinson, Matt Sharpe,* University of Rochester

In order to gauge the impurities entering the tritium fuel cycle (FC) in an ICF target chamber, we focus on residual target material left after a shot in the forms of ablation coating and gaseous species. The ablation coating, or material left on surface of chamber, was measured by placing sapphire wafers 26 cm away from the target. The gaseous species are measured via Residual Gas Analyzer (RGA) directly mounted on the ICF target chamber. The RGA provides mass information for the major gas species residual in the chamber. After being exposed to an ICF target implosion, the wafers will be separated into two testing groups: one to measure tritium activity and the other to measure surface changes. Tritium surface activity will be measured through etching, while surface changes will be measured by X-ray photoelectron spectroscopy and X-ray reflectivity. The goal is to understand the depth that tritium reaches, surface changes on the wafer, contaminants left on surface, and contaminants left in gas.

This material is based upon work supported by the Department of Energy [National Nuclear Security Administration] University of Rochester "National Inertial Confinement Fusion Program" under Award Number(s) DE-NA0004144; Data were produced by Battelle Savannah River Alliance, LLC under Contract No.89303321CEM000080 with the Department of Energy.

11:45am **FUS-WeM-16 FORGED-PFC High Heat Flux Facility Vacuum Vessel Design Status***, **Jared Tippens**, Ezekial Unterberg,* Oak Ridge National Laboratory

Oak Ridge National Laboratory has partnered with Type One Energy and The University of Tennessee Knoxville to build a world-class High Heat Flux Experimental facility focused on testing and qualifying fusion energy plasma-facing components (PFCs). This facility will utilize two converging high-power electron beams (~1.6 MW total) to heat test articles at fusion-relevant PFC power densities. Heat will be removed from the test articles via a dedicated helium loop with flowrates up to 1 kg/sec at a pressure of 8 MPa. Large components measuring over 1 m x 1 m x 0.5 m can be tested in this facility, enabling sub-component PFCs to be evaluated under representative operating conditions for thermomechanical stress testing and qualification. There are currently no dedicated testing facilities in the Western Hemisphere with these capabilities. Therefore, the FORGED-PFC facility will fill this needed capability and operate as a community resource to accelerate domestic fusion research and deployment of fusion energy systems.

For effective high heat flux testing at this scale, experiments must take place in a vacuum environment at pressures on the order of $1x10^{-4}$ mbar and lower. The two electron beams will be mounted to a relatively large vacuum vessel (~ 10 m^3) that will also house the test articles during experiments. Robust thermal management of the vacuum vessel is required to dissipate up to 1.6 MW of heat from the inner walls, necessitating a jacketed design. In addition, numerous ports are required for diagnostics to enable reactor relevant measurements and support predictive modeling validation. To achieve target pressures and minimize downtime between experiments, a multi-stage roughing pump system is required in tandem with cryopumps. To optimize the layout of the electron beams and diagnostics relative to the test articles, a custom D-shaped vacuum vessel is envisioned for this application.

An overview of the FORGED-PFC project and its current status will be provided, along with design details of this high-performance vacuum vessel and associated pumping system.

Note: This work is supported by the U.S. Department of Energy contract DE-AC05-00OR22725. The views and opinions expressed herein do not necessarily reflect those of Oak Ridge National Laboratory.

12:00pm **FUS-WeM-17 Materials and Deposition Technologies Underlying Target Fabrication for Inertial Confinement Fusion***, **Salmaan Baxamusa**, Sean Hayes, Liam Sohngen, Juergen Biener,* Lawrence Livermore National Laboratory*; Christoph Wild, Tobias Fehrenbach,* Diamond Materials GMBH, Germany*; Casey Kong, Mark Ratledge, Anthony Allen,* General Atomics*; Corie Horwood, Parminderdeep Singh, Alec Schwartz,* Lawrence Livermore National Laboratory*; Loosineh Aghaian,* General Atomics*; Bernard Kozioziemski, Laura Kegelmeyer, Gabrielle Hamza,* Lawrence Livermore National Laboratory*; Nancy Spafford,* Lawrence Livermore National Lab

The demonstration of laser-based inertial confinement fusion (ICF) ignition on the National Ignition Facility (NIF) in 2022 was a triumph of integrated physics and decadal science. This experiment, and the subsequent experiments building up to energy gains >4, are to date the only experimental demonstrations of fusion exceeding the Lawson criterion in the laboratory. Yet underlying this physics result was a complex blend of materials science and vacuum deposition needed to create and field an igniting target. This presentation will cover two aspects of the igniting physics package where vacuum deposition plays a key role: the hollow, spherical fuel capsule that implodes and the cylindrical hohlraum that provides x-ray drive for the implosion.

The capsule used in ignition experiments is CVD-deposited nanocrystalline diamond deposited. This talk will describe improvements to fabrication including machine-learning based diagnostics, Monte Carlo modeling, integrated experiments, and infrared reflectometry to improve coating thickness uniformity to > 99.8% on non-planar, non-stationary substrates.X-ray computed tomography was used to confirm that improvements to coating uniformity did not create additional internal defects.Finally, by controlling the composition along the growth axis we created capsules that stabilized the implosion and resulted in a record gain of >4. We will briefly mention materials science challenges of other capsule materials including boron carbide and glow discharge polymer.

The cylindrical hohlraum that contains the imploding capsule is a high-atomic number material (gold and uranium) that converts incoming laser light into a spherical bath of x-rays that drives the capsule implosion.Despite decades of study, the complex oxidation pathways of uranium are incompletely understood.We observed catastrophic oxide spalling on depleted uranium hohlraums that requires both an oxidant during processing and an environmental trigger. An experimental study confirmed that removing one or the other effectively increased the shelf-life of these components from ~weeks to ~months or years, eliminating a major cause of late-stage target failures for NIF.

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

Materials for Fusion Fuel Containment

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

2:15pm **FUS-WeA-1 MAX Phase Materials for Hydrogen Permeation Barriers***, **Eric Vogel**, Moses Nnaji,* Georgia Institute of Technology*; Dale Hitchcock,* Savannah River National Lab **INVITED**

Hydrogen isotope permeation is a critical materials challenge for fusion and other nuclear technologies, where tritium loss, inventory control, and degradation of structural alloys must be managed under high temperature and radiation. Conventional ceramic barriers such as Al_2O_3 can provide large permeation reduction factors, but brittleness, thermal-expansion mismatch, radiation sensitivity, and high processing temperatures limit integration with engineering alloys. This talk will present Ti_2AlN MAX phase thin films as radiation-tolerant hydrogen isotope permeation barriers. MAX phases combine ceramic stability with metallic transport, radiation damage tolerance, oxidation resistance, and the ability to form protective alumina scales.

The presentation will focus on low-temperature vapor deposition routes for MAX phase Ti-Al-N coatings, especially two-stage reactive magnetron sputtering. Ti-Al-N precursor films are deposited near ambient temperature and then annealed to drive solid-state reactions and Ti_2AlN crystallization. By controlling precursor architecture, composition, and modulation period, the formation pathway can be shifted from conventional high-temperature synthesis toward substrate-compatible processing. XRD, DSC-TGA, STEM show that Ti/AlN and TiN/TiAl multilayers follow distinct reaction pathways, with Ti/AlN enabling Ti_2AlN formation at lower temperatures. Trace Ti_2AlN was observed after extended annealing at 475 °C, an unusually low synthesis temperature for Ti-based MAX phase films by sputtering, while higher-temperature anneals produced crystalline Ti_2AlN coatings for permeation testing.

Deuterium permeation measurements on coated Grade 91 steel demonstrate substantial flux reductions relative to uncoated steel. The best Ti_2AlN coatings produced permeation reduction factors from tens to more than 1000, depending on temperature, pressure, film texture, and surface condition. Strong basal texture, improved crystallinity, and reduced high-angle grain-boundary content were correlated with improved barrier performance, consistent with anisotropic diffusion in layered MAX phase structures. Non-ideal behavior, including surface-limited dissociation, trapping, residual Ti-rich layers, oxide/carbon-rich surfaces, and pressure-induced flux changes, will be discussed. Together, these results identify processing-structure-property relationships that can guide radiation-hard, substrate-compatible MAX phase permeation barriers.

2:45pm **FUS-WeA-3 Atomic Layer Deposition on Novel Fusion-Relevant Substrates***, **Zachary Robinson**,* The Laboratory for Laser Energetics, The University of Rochester*; Soren Bentley,* UKAEA, UK*; Jeffrey Woodward,* NRL*; Alexander Kozen,* University of Vermont*; Thomas Cochran, Rashad Ahmadov, Joshua Ruby, Mark Wittman, Matthew Sharpe,* The Laboratory for Laser Energetics, The University of Rochester

Atomic layer deposition (ALD) is an ideal technique for deposition of films on arbitrarily-shaped containers such as those used to contain tritium for fusion applications. The substrates required for such containers generally differ from common ALD substrates, which have historically been primarily semiconductors. In this talk, I will give an overview of a custom ALD system built to deposit permeation barrier films on substrates and components relevant for various fusion and fusion fuel-cycle applications.

The primary benefit of our ALD system is the ability to deposit films on arbitrarily-shaped surfaces, such as the interior walls of tubing and canisters used to contain tritium. Initially, we deposited thermal ALD alumina films on both silicon wafers and planar copper foil substrates. Characterization with ellipsometry yielded ALD growth rates of ~1.1 Å/cycle for temperatures between 100 °C and 210 °C on Si witness samples. X-ray photoelectron spectroscopy (XPS) on both the Cu foils and Si substrates indicates stoichiometric Al_2O_3. To quantify the permeation reduction factor (PRF) of the of the ALD films, the permeability of deuterium through 25µm-thick Cu-foils coated with 10 nm of alumina was measured. We found that thin ALD films have a PRF of ~25 at permeation temperatures between 275 °C and 350 °C.

Finally, we discuss the difficulty of achieving a high quality ALD-stainless steel 316 (SS316) interface, which is complicated by the complex surface chemistry and the interactions with vapor-phase metalorganic species. The native oxide on SS316 substrates is reactive with TMA, and forms an AlOH layer, as evidenced by XPS measurements. To prevent this interfacial hydroxide formation, substrates were polished and conditioned in the ALD reactor. Following these steps, linear growth rates of 1.2Å/cycle were achieved. The ALD films on steel substrates resulted in orders of magnitude reduction in hydrogen permeation.

Current efforts include developing new materials in our ALD reactor, including a fundamental study of ALD bilayer and nanolaminate performance as a hydrogen-permeation barrier.

This material is based upon work supported by the Department of Energy [National Nuclear Security Administration] University of Rochester "National Inertial Confinement Fusion Program" under Award Number(s) DE-NA0004144.

3:00pm **FUS-WeA-4 Passivation of Stainless Steel for Tritium Service***, **Dale Hitchcock**,* Savannah River National Lab

Austenitic steels are generally preferred for hydrogen/tritium service due to their low hydrogen isotope permeability and relative resistance to hydrogen-isotope embrittlement. In applications where high purity gas is needed the surfaces in contact with hydrogen isotopes must be passivated to maintain gas composition. Passivation serves to minimize the surface features that catalyze isotopic exchange and to form a barrier against the ingress of impurities from the bulk. The passivation process generally includes an electropolishing step designed to minimize surface area by removing surface roughness and control surface chemistry followed by vacuum firing to both remove solubilized hydrogen from the bulk and also further modify the surface chemistry. Previously multiple vendors have supplied passivated components to the DOE, however, procuring "well" passivated components has become more difficult. This presentation will outline current efforts at SRNL to both evaluate the performance of vendor passivated material and also develop a robust passivation process inhouse.

3:15pm **FUS-WeA-5 Measurement of Hydrogen Isotopologues and Contaminant Species Within a Canister Coated with Alumina by Atomic Layer Deposition***, **Matthew Sharpe**, Thomas Cochran, Zachary Robinson,* University of Rochester

The interaction of tritium with pristine metallic surfaces catalyzes the formation of various hydrogen molecular isotopologues (HD, HT, DT, etc.). Additionally, because these surfaces have not been exposed to tritium, protium (H) native to the surface will enter the gas phase by isotope exchange with tritium. For many Fusion applications, this dilutes the gaseous tritium concentration, thereby contaminating the gaseous tritium with undesired isotopes. Furthermore, protium present within the bulk structure of the metal can act as a source of additional contamination. Finally, any trace organics present on the surface may also enter the gas phase as tritiated compounds. These organic molecules may poison other tritium technologies or become a health or environmental risk. Mitigating this contamination may be possible by use of atomic layer deposition (ALD) of low permeable materials.

In this work, we show the results of an aging study of a stainless-steel (type 316) canister coated with an alumina layer deposited on the interior surface by ALD. The canister was mounted in a custom ALD system designed for deposition on arbitrarily-shaped and fusion-relevant components. In a series of measurements performed to evaluate the deposition in a canister similar to those used in this study, a 10% variation in film thickness was observed between the inlet and outlet of the canister. Films of > 10 nm were used for this study, deposited at a temperature of 150°C.

Once the ALD-coated canisters were prepared, several similar canisters were prepared with pre-treatments including solvent surface cleaning and vacuum baking. After pre-treatment or ALD coating, the canisters are exposed to tritium gas. Assessment of the effectiveness of various treatments is done by periodically sampling the gas phase and measuring with gas chromatography (GC). The GC method utilized allows for simultaneous injection of a sample into two columns. One column allows for measurement of common impurities (CH_4, CO_2, CO, O_2, N_2, Ar), while the second allows for discrimination of hydrogen isotopologues (H_2, HD, HT, D_2, DT, T_2).

This material is based upon work supported by the Department of Energy [National Nuclear Security Administration] University of Rochester

"National Inertial Confinement Fusion Program" under Award Number(s) DE-NA0004144.

3:30pm **FUS-WeA-6 Native Protium Content in Stainless Steel Pipework***,* ***James Pittard****, Kyle Shipman, Zhiheng Wu, Tshepo Mahafa, Rosemary Brown, Gowri Karajgikar, Iryna Bennett,* UKAEA, UK

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 1018 H atoms g-1 (101 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.

Vacuum Technology
Room 321 - Session VT+FUS-WeA

Vacuum Technology Fusion

Moderators: Freek Molkenboer, TNO Science and Industry, the Netherlands, **Dr. Alan van Drie**, TAE Technologies

4:15pm **VT+FUS-WeA-9 The "Wall" in Fusion Reactors***,* ***David Ruzic****,* University of Illinois at Urbana-Champaign **INVITED**

Due to the recent invention of flexible high-field super-conducting magnets which can sustain fields up to 20 Tesla, commercial magnetic fusion energy may soon be a reality.These fields will contain a thermonuclear plasma that could have equivalent heat loads to the surface of the sun – some 60 MW/m^2.Nothing can withstand that power load, so much effort has gone into designing systems that will "only" experience on the order of 10 MW/m^2.That wall material is not allowed to contaminate the plasma or the fusion reaction will stop.This issue of the plasma-facing-component is one of the most serious engineering challenges.However, there are more challenges.A fusion device must make more tritium than it uses, and that includes tritium permeation.A breeder blanket will surround the plasma which will contain lithium.Neutrons from the D-T reaction will hit lithium and make more T.That tritium needs to be recovered and contained.Finally the structure of the fusion device must be able to withstand enormous transient loads from magnetic excursions and not fall apart due to transmutations and atomic displacements from the copious neutron flux.And, by the way, the whole system has to e vacuum tight.This talk will explain all of these challenges and describe the leading solutions and largest uncertainties.

4:45pm **VT+FUS-WeA-11 Overview, Construction Status, and Engineering Challenges of the Vacuum Systems for the Material Plasma Exposure eXperiment (MPEX)***,* ***Jonathan Perry****, Aftab Hussain, Adam Aaron, Ted Biewer,* ORNL

The Material Plasma Exposure eXperiment (MPEX) is under construction at Oak Ridge National Laboratory. MPEX will be a first-of-its-kind steady-state linear plasma device that will enable neutron-irradiated materials to be exposed to fusion reactor-relevant divertor plasma conditions for the study of plasma-material interaction. This facility will be capable of generating high heat fluxes and high ion fluences to test fusion divertor prototypic plasma-facing materials at steady-state for up to 10^6 seconds with in-situ diagnostics and magnetic fields up to 2.5 T. This presentation will provide an overview and construction status of the MPEX vacuum systems that promote efficient heating of the plasma, create an environment for prototypic divertor plasma conditions, enable in-situ diagnostics, and maintain vacuum during transportation and post-exposure diagnostic examination of the tested material. Due to the operating environment and performance requirements of MPEX, many engineering challenges came up during the design, fabrication, and assembly phases of the MPEX vacuum systems. This presentation will also discuss the lessons learned and implemented, as well as the knowledge leveraged from the ITER project and other facilities around the world, to develop, manufacture, and assemble the vacuum systems for MPEX. Unique cleaning and assembly techniques to ensure cleanliness is maintained throughout the assembly and installation process will also be discussed in this presentation.

This work was supported by the Oak Ridge National Laboratory managed by UT-Battelle, LLC for the U.S. Department of Energy under Contract No. DE-AC05-00OR22725.

5:00pm **VT+FUS-WeA-12 Miniaturized, Fully Shielded High-Voltage Vacuum Feedthroughs from 50 kV to 300 kV***,* ***Moein Borghei****,* Avalanche Energy

High-voltage vacuum feedthroughs rated above 100 kV are rarely available as commercial off-the-shelf products, and those at 100 kV or below are often oversized and unshielded, i.e., have exposed electrodes at high voltage. These gaps limit researchers and engineers building fusion machines, electron guns, X-ray sources, ion beam systems, and other high-voltage vacuum platforms, who must either tolerate poor component fit or resort to lengthy custom procurement cycles.

To meet the diverse high-voltage needs of Orbitron fusion machines [1], we developed a suite of four fully shielded, CF-flanged feedthroughs. These models -the Thresher, Bigeye, Mako, and Hammerhead- span a wide voltage range from 50 kV to 300 kV. All designs utilize low-outgassing ceramic insulators, such as aluminum oxide and MACOR. The ceramic component is hermetically sealed to standard stainless steel conflat flanges, with leak rates lower than 1×10^{-9} atm·cc/s of He. The feedthroughs are fully shielded, eliminating exposure of high-voltage surfaces on the outside, thus enhancing integration within compact systems and operator safety.

The primary design objective was to minimize the footprint while ensuring reliable, steady-state operation. Mako and Hammerhead are within 15 cm and 20 cm flange sizes, respectively, both half the volume of the designs provided in the literature for the voltage class [2]. Similarly, Bigeye transmits 100 kV through a 7 cm flange compared to the 15 cm flange of the COTS feedthrough. Thresher does the same for 50 kV in a 3.4cm instead of 7 cm.

In this talk, we present experimental results and design methodology for insulator geometry, electrode designs, and electric-field stress mitigation that simultaneously enable achieving the rated voltage while maintaining hermetic seal integrity. We also present measured performance metrics across these feedthroughs, including dark current, bakeout impact, role of contaminants, and the conditioning process.

[1] M. Borghei, M. Vorenkamp, R. Langtry, and B. Riordan, "An Overview of Vacuum Insulation Technology for the Orbitron Fusion Device," in *2025 31st International Symposium on Discharges and Electrical Insulation in Vacuum (ISDEIV)*, Chengdu, China: IEEE, Sep. 2025, pp. 1–5. doi: 10.1109/ISDEIV60861.2025.11256035 [https://doi.org/10.1109/ISDEIV60861.2025.11256035] .

[2] M. Borghei, M. Vorenkamp, N. Van Moon, R. McMullen, R. Langtry, and B. Riordan, "Hammerhead: a compact 300 kV vacuum bushing," *Nature Communications*, vol. 16, no. 1, p. 11391, Dec. 2025, doi: 10.1038/s41467-025-66194-w [https://doi.org/10.1038/s41467-025-66194-w] .

Materials and Innovations for Fusion Energy
Room Ballroom A - Session FUS-ThP

Materials and Innovations for Fusion Energy Poster Session

FUS-ThP-1 Experiment Setup of a Testbed Facility to Characterize Tritium Storage Materials, ***Simon Niemes***, *David Beisiegel, Nicolas Bekris, Noah Ellenbogen, Robin Größle, Ralph Lietzow, Immanuel Müller, Florian Priester,* Karlsruhe Institute of Technology (KIT), Germany*; Tim Teichmann,* Kyoto Fusioneering (KF), Germany

A key requirement for the transportation, storage and processing of tritium is the availability of storage devices compatible with tritium. Metal hydride storage bed, usually based on depleted uranium, are commonly used due to the low equilibrium pressure at room temperature and the efficient tritium release at temperatures above 400°C. However, the use of uranium requires a dedicated licencing and additional safety measures due to the pyrophoric nature of the storage material. Therefore, research efforts into alternative materials like ZrCo based alloys are required. At the Tritium Laboratory Karlsruhe (TLK), a new dedicated setup is currently being commissioned to assess the storage properties with high accuracy and using an inventory of up to 3 grams of tritium. This setup will be able to perform thermodynamic and kinetic measurements as well as investigations of disproportionation and regeneration behaviour for different getter bed materials

This contribution gives an overview of the working principles and status of the experiment and the considerations for the use with tritium.

FUS-ThP-3 Development of a Method for Hydrogen Species Analysis in Fusion Breeder Materials Using ICP-MS, ***Clemence Mont***, UKAEA, UK

The UK Atomic Energy Authority is investigating breeder material technologies through the LIBRTI (Lithium Breeding Tritium Innovation) programme to support tritium production for Fusion Energy. Tritium is a scarce resource around the world and one of the key fuels for Fusion with deuterium. The Fusion sector must develop breeding technologies to transform lithium into tritium using the neutrons generated during the fusion of deuterium and tritium. This will create a tritium fuel cycle where tritium is created as it is used for the fusion reaction.

The LIBRTI programme is investigating different breeder materials such as FLiBe, LiPb and liquid lithium among others. A facility will be built in the coming years on the Culham Campus in Oxfordshire, UK, to test the materials in a large-scale mock-up experiment. Within the LIBRTI programme, a project called CLIO is looking at developing an analytical method to determine the concentration of tritium in the breeder materials using ICP-MS (Inductively Coupled Plasma – Mass Spectrometry). Developing analytical techniques for tritium accountancy is a priority to ensure process control, safety, environmental protection and security around a fusion power plant.

The CLIO project is split into 4 phases. The first 1 started in June 2026, and its aim was to develop a method for hydrogen and deuterium analysis as surrogate to tritium. Hydrogen is famously difficult to analyse with an ICP due to its high ionisation energy, the development of a novel method would allow the instrument to be used in the Fusion sector. Phase 2, started in October 2026, introduces breeder materials as a sample. The detection of hydrogen might suffer from matrix effect coming from the different type of materials, assessing each of them is important to ensure accuracy of the results. Phase 3 starting in January 2027 will look at simplifying the sampling technique by adding the Laser Ablation – LIBS (Laser Induced Breakdown Spectrometry) system to the ICP-MS. While using the ICP-MS on its own necessitates the dissolution of the sample, the LA-LIBS add-on will allow surface analysis. The LA-LIBS-ICP-MS will be able to do bulk and surface analysis. The final phase planned later in 2027 will introduce tritium to the system. This step will ensure that the system can analyse all hydrogen isotopes and that it can be used in the LIBRTI programme as a tritium detection system.

Results from the method development carried out in phase 1 as well as preliminary data from phase 2 are presented in the poster.

FUS-ThP-4 the Actively Pumped Open-Surface Lithium LOop (APOLLO) – a Liquid Lithium Plasma Facing Component Technology Testbed, ***Brady Moore***, University of Illinois Urbana-Champaign*; Daniel O'Dea,* University of Illinois at Urbana Champaign*; Zachary Nordan, David Ruzic,* University of Illinois at Urbana-Champaign

Liquid lithium is being explored as a plasma-facing component (PFC) for fusion devices, particularly in the divertor region. Compared with traditional solid materials like tungsten, a flowing liquid surface can better recover from extreme heat fluxes during transient events, continually renew itself, and is impervious to neutron damage. Lithium also has a low atomic number leading to a higher tolerable concentration in the plasma core. Additionally, lithium readily reacts with hydrogen isotopes leading to a low-recycling boundary that can improve plasma performance and flatten plasma temperature profiles. However, this same property creates a key challenge: lithium can retain radioactive tritium which has strictly monitored inventories. Future fusion reactors will aim to minimize total tritium inventory and therefore effective tritium separation from liquid lithium is critical.

The University of Illinois Urbana–Champaign (UIUC), in partnership with Tokamak Energy Ltd., has developed the Actively Pumped Open-Surface Lithium Loop (APOLLO) to develop technologies required for implementing liquid lithium PFCs. APOLLO features a circulating liquid lithium loop, a free-surface lithium plasma-facing component operating under a magnetic field, a hydrogen/deuterium plasma source or electron beam heating system, and an in-line distillation column for the removal of hydrogenic species. The PFC employs a computationally optimized distributor that ensures uniform delivery of lithium from an inlet pipe across a 7.5 cm-wide, additively manufactured refractory metal mesh within an open-surface flow channel. Lithium traverses the mesh at average surface velocities up to 10 cm/s, with mass flow rates reaching 12 g/s, while simultaneously interacting with an electron cyclotron resonance (ECR) hydrogen/deuterium plasma. Plasma properties have been characterized in-situ using an array of 16 Langmuir probes, a retarding field energy analyzer (RFEA), and actinometric spectroscopy. After plasma exposure, the lithium is directed to the inductively heated Hydrogen Distillation Experiment (HyDE), where it is thermally processed at temperatures up to 700 °C to remove hydrogenic species and other impurities before being recirculated through the loop.

In this work, hydrogen uptake in flowing liquid lithium exposed to an ECR plasma, as measured by a novel resistive impurity probe, is investigated as a function of lithium flow rate and plasma operating conditions. The efficiency of hydrogen thermal extraction is also quantified using a resistive impurity probe and thermal desorption analysis.

FUS-ThP-7 An Analysis of Radiation Tolerance of Parylene Coatings for Nuclear Energy Applications, ***Steven Larson***, *Alex Mings, Joshua Young, Ronald Goeke,* Sandia National Laboratories

Steven R Larson, Alex Mings, Josh Young, Ron GoekeSandia National Laboratories

Parylene is a chemical vapor deposited (CVD) polymer coating widely used in the electronics industry due to its excellent chemical resistance and dielectric properties. Recently, Parylene has also attracted interest as a protective coating for components in nuclear reactor systems, where materials are often exposed to highly corrosive environments, elevated temperatures, and high radiation. Despite this interest, relatively little is known about how the different Parylene polymers response to ionizing radiation. Additionally, the extent to which radiation tolerance varies among the different types of parylene remains largely unexplored. While some data exists for parylene C, D, and N very little data has been published on the different parylene F polymers (VT-4 and AF-4) due to their difficulty in precursor fabrication. This talk will present efforts to determine the physical, chemical, optical, and electronic changes that occur in Parylene C, D, N, VT-4, and AF-4 following exposure to lifetime-relevant gamma and neutron irradiation. We will discuss the differences between the various parylene polymers and how to cost optimize for extremely expensive precursors. Sandia National Laboratories is managed and operated by NTESS under DOE NNSA contract DE-NA0003525 SAND2026-21514A

FUS-ThP-8 Torion USA Multi-Gram Tritium User Facility, ***Cody Fagan***, *Mike Koch,* Torion USA*; C.R. Shmayda,* Torion Plasma, Canada*; Walter Shmayda,* Tritium Solutions

The opening of the Torion USA multi-gram tritium facility marks a milestone in advancing fusion fuel cycle (FFC) science, engineering, and workforce development in the US. Designed as a flexible, user-oriented platform for experiments, prototyping, and applied training, the facility establishes a

new private capability for hands-on engagement with FFC technologies at scales relevant to fusion demonstration systems.

The facility is equipped with a suite of state-of-the-art systems supporting tritium processing, accountancy, isotope separation, permeation, materials compatibility testing, and detritiation. Multiple modular glovebox systems and experimental stations are available to support tritium process equipment incorporating high technology readiness level (TRL) components. These systems are intended to demonstrate integrated FFC operations while also providing a place for deploying and validation of our next-generation technologies that are relevant to Fusion Pilot Plants (FPPs). By enabling both operations and technology advancemet, the user-facility is designed to accelerate innovation in FFC technologies while reducing barriers to access for emerging fusion companies and research partners to test developed concepts with tritium.

A central mission of the facility is the development of a highly trained fusion workforce capable of safely operating our advanced tritium systems. The laboratory will host immersive training programs that combine classroom instruction with direct hands-on operational experience using our glovebox systems and experimental stations. Participants will gain practical experience in tritium radiological safety, regulatory compliance, process control and maintenance of systems, and experimental methods essential for future fusion energy deployment. These programs are intended to cultivate the next generation of fusion scientists, engineers, operators, and safety professionals at a time of rapidly increasing demand across the fusion energy ecosystem.

FUS-ThP-9 Investigating Gaseous Impurity Production in a Baked UHV System*, Gabriel Wilber,* Tritium Solutions, Inc.*; E. Koukina, K. McCormack, A. Knaian,* Acceleron Fusion*; W.T. Shmayda,* Tritium Solutions, Inc.

Muon-catalyzed fusion offers a unique alternative energy source. In this approach muons are injected into a high-pressure gas target containing a deuterium-tritium mixture. When a muon attaches itselves to a DT molecule, the probability for the DT molecule to fuse increases significantly. This event generates energy, creates a 4He atom, and releases the muon to attach itself to another DT molecule over its 2.2 µs mean lifetime. However, if the muon interacts with a gaseous impurity, it can no longer participate in catalyzing fusion reactions. This paper investigates on the type and quantity of impurities produced when D2 is injected into a heated 316-stainless steel ultrahigh vacuum system.

Author Index

Bold page numbers indicate presenter