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Improving TEA with COFFEE and CREAM: Incorporating Risk and Uncertainty into Techno-Economic Analysis

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Monday Plenary - Andrew Lo, Kresge Main Theater (Building W16, upstairs), June 23, 2025, 9:00 AM - 10:00 AM

Techno-economic analysis (TEA) plays a critical role in evaluating the economic viability of deep-tech initiatives by incorporating scientific and engineering considerations into traditional economic and business modeling. In this talk, Prof. Lo will propose a general framework for TEA that is versatile enough to be applied across a wide range of industries and ventures, with special emphasis on Capital Outlays For Funding Early Enterprises (COFFEE) and Corporate Risk Estimation And Management (CREAM). While this framework is broadly applicable, its focus is on technologies that are First-of-a-Kind (FOAK), as they present unique challenges in terms of uncertainty and risk. An illustrative example will be provided to show how COFFEE and CREAM can improve TEA for various deep-tech ventures including biotechnology and nuclear power.

Commercialization and Industrialization of Fusion Energy by Organization of Supply Chain

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Monday Parallel 1a - Commercialization, Kresge Main Theater (Building W16, upstairs), June 23, 2025, 10:30 AM - 12:00 PM

The current fusion development is considerably led by private sector. Both public and private entities aim at the First Of A Kind fusion energy systems such as DEMOs or Pilot Plants, however the technical strategies to follow will be rather different, because privates consider immediate deployment of their technical achievements and products into the market. For example, while the technical success of ITER magnet fabrication was not followed by the similar commercial products, the current private High Temperature Superconducting magnet market led by private sector expects continued production to expand its market following first successful generation machines. Because current fusion programs are oriented the industrialization of fusion and encompass organization of supply chain as a major element, integration of the fusion energy system will have to be developed and verified to the Technical Readiness Level to be confident to provide as commercial products. Fusion energy market is already being formed and competition for the share has started even before its first sales of energy products.

Particular technical difficulties are identified as important in the field of blanket, material, fuel cycle, energy conversion and nuclear technology. Because most of the projects currently going on or planned emphasize plasma performance toward fusion reaction, these nuclear and plant technology are concerned to be left as the major critical paths toward commercialization. This talk will overview the current status of the fusion technology developments focused on private initiatives, and possible early stage attempts for DT burning, blanket and energy technology. Typically the FAST (Fusion by Advanced Superconducting Tokamak) project and its initial design led by Japanese private sector will show a good example of the expected function as a platform to develop, evaluate and improve the maturity of fusion energy technology that will be essential for power plants. It is expected to start its operation in mid 2030 to generate the reactor relevant volumetric neutron flux with facing burning DT plasma for technically meaning time scale of 1000 seconds. Advanced low aspect ratio tokamak configuration with high temperature superconducting magnets is applied. Breeding blanket with coolant will be equipped with fully functional thermal energy cycle and tritium extraction process to verify their function under relevant condition. Power conversion system will generate electricity as well as other energy applications such as hydrogen production from high grade heat from blankets at the level of 100MW.

Leveraging public-private partnerships to unleash fusion energy development in the U.S. and beyond

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Monday Parallel 1a - Commercialization, Kresge Main Theater (Building W16, upstairs), June 23, 2025, 10:30 AM - 12:00 PM

American dominance in fusion energy will be predicated on the ability to build effective bridges between public and private sectors. The Office of Science (SC) Fusion Energy Sciences (FES) is building a portfolio of exciting opportunities bringing these sectors together. The Milestone-Based Fusion Energy Development program, INFUSE, Private Research Facility (PFR), are among these key elements. Furthermore, to best serve the interest of the American taxpayer and leverage the investment of the uniquely vibrant U.S. private sector in fusion energy, SC FES released a Request for Information on a Public-Private Consortium Framework (PPCF) and now branded as Fusion BRIDGE (Bringing Regional Investments to Develop & Grow a US fusion Engine) pilot to leverage public-private partnerships to support innovation in fusion energy technology development. Details of these elements and how it aligns with the SC FES mission to support a competitive fusion power industry will be presented.

Spain as showcase for electricity grid needs

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Monday Parallel 1a - Commercialization, Kresge Main Theater (Building W16, upstairs), June 23, 2025, 10:30 AM - 12:00 PM

In Spain, the demand for grid access is reaching very high levels due to the high appetite for electrification of existing production processes, new industry looking to invest in Spain due to the possibility to sign long term PPA contracts and investments in data centers, among others. Access requests reach levels that could triple or quadruple all the power currently connected to the Transmission and High and Medium Voltage Distribution networks.
The grid is becoming a bottleneck for the development of all access and connection requests.
Description of all the initiatives that could be developed to maximise the use of the network and to ensure a development that would allow to accept the increased demand.

First operation results of the new upper Divertor in ASDEX Upgrade

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Monday Parallel 1b - Magnets and Cryogenic Systems I, Sala de Puerto Rico (Building W20 Room 202), June 23, 2025, 10:30 AM - 12:00 PM

ASDEX Upgrade is a midsize tokamak that has been operational since 1991, serving as a key platform for fusion research with a total heating power up to 32 MW. Following a two-year maintenance period, ASDEX Upgrade resumed operations in November 2024. During this upgrade phase, a new upper divertor was installed to enhance the machine's flexibility. The modifications included the installation of an upper cryopump capable of extracting helium using charcoal, a new flat divertor allowing bi-directional magnetic helicity, and two concentric coils designed to enable alternative divertor configurations for improved exhaust handling.
The reassembly of the machine was completed within the planned timeframe autumn 2024. However, due to time constraints, the electrical connection of the newly installed coils is scheduled to be operational in early spring 2025. Consequently, the initial commissioning phase was conducted without powering these coils. Despite this limitation, it was still possible to characterize the behaviour of the upper divertor.
A significant initial finding is that, in the first upper single-null configurations, no leading-edge heat loads were observed on the outer divertor tiles, which feature a flat profile—even in the presence of disruption events. A more detailed assessment of the tile conditions will be available following the scheduled machine inspection during the summer maintenance period.
Another interesting result comes from the first measurements characterizing the upper cryopump. The cryopump consists of nearly seven identical modules, connected in series, toroidally distributed, and positioned between the vacuum vessel and the inner divertor cooling plate. Each module comprises 4K cryopanels coated with activated charcoal, housed within an 80K shield and a chevron baffle. The upper cryopump is designed for a minimum effective pumping speed of 21 m^3/s for D_2, but this value has been slightly exceeded during operation. In fact, depending on the pressure, a value between 22 and 26 m^3/s was measured. Nevertheless, during plasma operation with neon, an additional unforeseen effect was observed. It was discovered that the charcoal absorbs neon, which is used to mitigate disruptions when the plasma current exceeds 800 kA. Once the helium panel has taken up neon, the upper cryopump releases it in subsequent shots. This issue appears to result from excessive heating of the LHe panel. Based on preliminary evaluations, three heating sources are causing this effect: the impingement of particles and their adsorption on the surface of the LHe panel; the ECRH stray field and the plasma radiation reaching the cryopump.
To ensure the continuation of the experimental campaign, a temporary workaround has been implemented. Meanwhile, potential solutions to address the cryopump issue are currently under evaluation. This paper will present the initial results from the commissioning of the new upper divertor and discuss ongoing efforts to optimize the system's performance.

Calculating the expected critical current for 3D coils of arbitrary shape made with VIPER-like HTS cable

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Monday Parallel 1b - Magnets and Cryogenic Systems I, Sala de Puerto Rico (Building W20 Room 202), June 23, 2025, 10:30 AM - 12:00 PM

Superconducting coils utilizing VIPER-like HTS cable for tokamaks and stellarators are being designed, built, and tested. In order to assess a coil's ability to operate at the desired current, and/or to evaluate possible degradation of a coil's performance due to manufacturing processes, it is necessary to calculate the expected critical current, Ic, of the coil. This is actually a difficult task, since the Ic of HTS tape varies non-linearly with B-field magnitude, B-field angle to the tape plane, and temperature. VIPER-like cables consist of stacks of HTS tape, and even though the cross-sectional area of a stack is typically only a few mm2, the near-field generated by the current in the stack is high enough to drastically reduce the Ic of the tapes within the stack. Therefore an accurate calculation of the expected Ic at any location along the coil requires modeling each stack of HTS tape as a dense array of current filaments, and the geometry of these filaments must reflect the geometry of the tape stacks, which wind helically around the cable axis with a twist pitch of order 100-200 mm. At high currents the far-field generated by the non-local parts of a coil can also reduce Ic at the location of interest, and therefore must also be modeled as a set of current filaments, which can be of lower spatial resolution, but which must reflect the geometry of the full coil. For irregular coil shapes, this entails using specialized 3D coordinate systems, such as Frenet-Serret or related frames of reference.

A code to calculate the expected Ic for arbitrarily shaped 3D coils using VIPER-like HTS cable has been developed at MIT PSFC. The code is specifically for coils operating in steady-state (such as for optimized stellarators). The basic physics assumption is that at any location along the coil, the E-field is uniform across the cable cross-section and equal to 1 uV cm (1e-4 V/m), which is the definition for HTS being at critical current. The code reads in a .csv file containing the xyz-coordinates of the cable centerline, and generates a dense array of current filaments representing the HTS tape stacks at the desired location of interest along the coil, and a low-resolution array of filaments representing the balance of the coil. The filaments reflect the actual geometry of the coil, and include the twisting geometry of the HTS tape stack in VIPER-like cables. The code iterates the current in each of the many current filaments at the location of interest, until each filament is at its Ic, which depends on the B-field magnitude at each filament, and the B-field angle to each filament. Due to the non-linearity of the problem, the calculation must be iterative, since changing the filament currents changes |B| and its angle, which changes the Ic of each filament, and so on. In addition to requiring the coil geometry as input, the code also requires the HTS tape characterization, Ic(B, angle, T), where T is the operating temperature of the coil.

Overview of magnet testing and development for SPARC

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Monday Parallel 1b - Magnets and Cryogenic Systems I, Sala de Puerto Rico (Building W20 Room 202), June 23, 2025, 10:30 AM - 12:00 PM

Commonwealth Fusion Systems (CFS) is advancing the SPARC tokamak, designed to achieve Q>1 and serve as a stepping stone toward ARC, the first fusion power plant. SPARC's high-field, compact tokamak design relies on groundbreaking High Temperature Superconducting (HTS) magnets, including Toroidal Field (TF), Poloidal Field (PF), and Central Solenoid (CS) coils. These magnets utilize two architectures: HTS Non-Insulated, Non-Twisted (NINT) technology for cryostable steady-state operation and HTS Cable technology for pulsed, high-field applications. Following successful R&D and demonstration campaigns through the TFMC and CSMC programs, CFS has established a comprehensive manufacturing and qualification pipeline to ensure the reliability of every magnet. This includes material qualification at low temperature and high field, sub-unit testing of pancake/layer performance at 77 K, and full-coil cryogenic testing at SPARC operating conditions. These tests de-risk the SPARC magnet systems and validate their readiness for integration, with all facilities designed for scalability to support SPARC's construction timeline.

Simplified Multiphysics Models for Quench in Non-Insulated Coils and Implications for Fusion Power Plants

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Monday Parallel 1b - Magnets and Cryogenic Systems I, Sala de Puerto Rico (Building W20 Room 202), June 23, 2025, 10:30 AM - 12:00 PM

Non-insulated (NI) magnets made of high-temperature superconducting tapes are of interest for a variety of different magnet applications, such as in the toroidal field magnets of future fusion reactors and for in-field critical current testing of HTS tapes. One of the primary reasons for this is the potential ability for NI coils to passively protect themselves against damage during a rapid global loss of superconducting behavior, known as a quench. Some prior tests of NI coils have demonstrated robust self-protection against thermal quench damage, but this does not appear to be universal; for example, during the intentional quench test of the SPARC Toroidal Field Model Coil (TFMC), the coil incurred significant thermal damage [1]. Consequently, the specific design and operational space in which NI coils are intrinsically self-protected against thermal quench damage is not well understood. To explore this self-protection regime and identify the primary coil design and operating conditions that will enable self-protection, a variety of multiphysics quench models of NI coils have been developed. Chief among these is a self-consistent axisymmetric 2D model of sudden-discharge in NI coils that was built using COMSOL Multiphysics [2]. This model has been validated against, and shows good agreement with, experimental sudden-discharge tests of NI and metal-insulated (MI) coils [2]. In addition, a slab model of individual conductor turns was developed and coupled to this 2D model to study how non-uniform critical current and current-sharing temperature within conductor turns impact the thermal and electromagnetic evolution of quench in non-axisymmetric coils, such as tokamak toroidal field coils. This coupled 2D-Slab model has been used to model the full winding pack of the SPARC TFMC, and validation against the experimental results of the TFMC test campaign is currently underway. This presentation will discuss the details of the developed quench models, validation of the models (including against the SPARC TFMC), and preliminary exploration of the passively-safe quench space and potential implications for the design and operation of NI coils for a variety of applications, including fusion energy.

[1] Z.S. Hartwig et al, “The SPARC Toroidal Field Model Coil Program.” IEEE Trans. Appl. Supercond., no. Special Issue on the SPARC Toroidal Field Model Coil Project, 2024.
[2] D. Korsun et al, “Simplified Multiphysics Models for Open-Circuit Quench in Non-Insulated and Metal-Insulated Superconducting Magnets.” IEEE Trans. Appl. Supercond. ASC 2024, submitted Sep. 2024, revised Dec. 2024, accepted Jan. 2025.

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Fusion Fuel Rebalancing Using Thermal Cycling Absorption Process

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Monday Parallel 1c - Tritium, Fueling, Exhaust, and Vacuum Systems I, Kresge Little Theater (Building W16, downstairs), June 23, 2025, 10:30 AM - 12:00 PM

Many fusion power concepts under development are based on using fusion of deuterium and tritium. One of the hurdles pertaining to the exploitation of this reaction is a necessity for continuous separation of hydrogen isotopes to be recycled to the fusion reactor core due to a very low fuel burn off (~ 1%). As shown in the figure below, a constant fuel rebalancing is required for reliable and sustainable fusion reactor operation (see blocks highlighted in green). There are a number of methods to separate tritium and deuterium with different technology maturity levels, amongst which the Thermal Cycling Absorption (TCA) process is considered a viable contender. Canadian Nuclear Laboratories (CNL) have developed several experimental systems for hydrogen isotope separation based on this technology, and continue improving, advancing, and scaling up this process. A substantial amount of data has been collected over the years, and the latest experimental results of hydrogen isotope separation (i.e., deuterium and protium) in the CNL's TCA system will be presented. Insights into the commissioning of a novel tritium compatible system, as well as a comparison between simulation results from a numerical model of the TCA process developed by CNL and experimental data, over a wide range of experimental conditions, will be provided.

Simplified schematic of a fusion fuel cycle

DT muon-catalyzed fusion experiments

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Monday Parallel 1c - Tritium, Fueling, Exhaust, and Vacuum Systems I, Kresge Little Theater (Building W16, downstairs), June 23, 2025, 10:30 AM - 12:00 PM

Acceleron Fusion Inc., under DoE ARPA-E funding, is exploring the efficacy of deuterium-tritium muon-catalyzed fusion at high temperatures (up to 1500K) and high pressures (up to 5 GPa) at the Paul Scherer Institute (PSI) in Switzerland. The objective of these studies is to guide the design concepts for power generation. To support this effort, tritium infrastructure has been installed at PSI and the Acceleron deuterium-deuterium muon experimental station has been upgraded to deploy 4000 Ci of tritium safely. This effort comprises the construction of a tritium gas handling system installed inside a glovebox. Helium is used as the cover gas inside the glovebox. This gas is continuously purified using a secondary enclosure cleanup (SEC) system to remove air permeating through the gloves and any tritium that has escaped from the primary process loop. Vacuum effluents from the gas handling system are treated to capture tritium from the remaining gas is discharged to the environment.

The primary process stainless-steel loop consists of a depleted uranium storage bed (DU bed), a 15 K cold finger, a palladium/silver permeator and a circulating pump. Loop contents are measured using pressure transducers and tritium process monitors. DT gas is transferred from the DU bed to the cold finger, expanded into and assayed in a 1-liter tritium monitor, and passed through the palladium - silver permeator to a diamond anvil cell held at 19 K. The DT gas entering the diamond anvil cell, stripped of decay helium and all other impurities, is liquified before the cell is closed to generate pressures as high as 930 MPa.

This presentation will outline the infrastructure installed at PSI and the key elements of the tritium gas handling system, describe the emission mitigation equipment and summarize the effectiveness of these packages during the week long execution of the experimental campaign.

Emerging physics and technological insights from the ASDEX Upgrade shattered pellet injector project

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Monday Parallel 1c - Tritium, Fueling, Exhaust, and Vacuum Systems I, Kresge Little Theater (Building W16, downstairs), June 23, 2025, 10:30 AM - 12:00 PM

The main goal of the ASDEX Upgrade (AUG) Shattered Pellet Injector (SPI) project - a collaboration between IPP, the ITER Organization, and EUROfusion - is to enhance the physics basis for protecting fusion reactors from large heat and particle loads by characterising the impact of the size and speed distribution of SPI fragments on disruption dynamics. The AUG SPI is one of the most flexible SPI systems. It has three independent guide tubes and pellet generation cells sharing a single cryocooler. The supported pellet materials include argon, neon, protium, deuterium, and mixtures of neon and deuterium. Barrel heaters allow for the fine control of pellet size and launch velocities between 70 and 800 m/s were achieved.

The three heads installed in the tokamak were chosen after an extensive characterisation of a dozen prospective SPI geometries. During the laboratory commissioning we have optimised the injector hardware, developed hundreds of automated pellet generation and launching recipes, and recorded more than 1300 ultra high speed videos of pellet fragmentation. I will discuss the technical and project management challenges of realising and operating such a flexible SPI system; as well as the computer vision and machine learning methods employed in the analysis and design.

We executed over 200 plasma discharges to optimise SPI material assimilation, heat load mitigation, and radiation asymmetries; and to generate a diverse dataset for model validation. Our results indicate that rectangular shatter heads with mitre bends lead to more reproducible fragment sprays. We observed pellet recipes leading to larger and faster fragments penetrate deeper and assimilate better in the plasma. Neon doping of less than 1% in majority deuterium pellets was observed to improve material assimilation. The total fraction of plasma energy radiated, and its spatial asymmetry, show strong dependence on the pellet composition (total number of neon atoms) but complex sensitivity to the shattering geometry. The results are corroborated by both 1D reduced kinetic and 3D MHD simulations. The AUG experimental and modelling results allow for a better design of SPIs for different fusion devices.

Tritium retention and protium ingress from stainless steel components during a fusion pilot plants commissioning phase

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Monday Parallel 1c - Tritium, Fueling, Exhaust, and Vacuum Systems I, Kresge Little Theater (Building W16, downstairs), June 23, 2025, 10:30 AM - 12:00 PM

A fusion pilot plant's tritium startup inventory defines the amount of tritium required on site to fill all processing components and make up for losses like decay, burn up, and diffusion. During this commissioning phase tritium diffusion and protium ingress have the potential to impact tritium startup inventory. Several components of the fuel cycle for example: storage tanks, piping, distillation columns, and pumps, will be made of stainless steel. The surface of stainless steel is an ideal place for water layers to form when exposed to atmospheric humidity. This layer absorbs tritium and deuterium through isotope exchange with protium, thus causing protium release and tritium/deuterium diffusion into stainless steel. In this work, protium ingress and tritium uptake in different components of the fuel cycle are investigated by using the quantitative tritium migration model that was developed by Dr. Sharpe at the University of Rochester. This model is based on Fickian diffusion, where water layers can be added to a metal surface. Preliminary results indicate that the amount of protium entering the fuel cycle for a mile of quarter inch piping is much smaller than the amount of tritium uptake by the pipe.

Advancement in the DTT First Wall design using additive manufacturing

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Monday Posters 1, Lobdell (Building W20 Room 208), June 23, 2025, 10:30 AM - 12:00 PM

The design of the DTT First Wall (FW) is under finalization, paving the way to the upcoming procurement stage. Recent qualification activities have identified the selective laser melting (SLM) technology as a highly promising additive manufacturing method for the steel-based plasma-facing components that form the FW. This approach offers enhanced design flexibility and mechanical properties. Additionally, the most relevant loading conditions were listed as an input for the engineering design. These include the deposition of thermal power during plasma operation and the generation of electro-magnetic loads during disruption events.
This contribution presents a refined configuration of a FW module, developed after extensive design optimization and simulation effort. The design was carefully adapted to meet the specific requirements of SLM manufacturing technology, ensuring compatibility with large-scale production. Subsequently, a detailed assessment of structural stability against the defined loading conditions was performed. These results provide the groundwork to proceed and conclude the engineering design activities.

The NSTX-U mission to bridge the spherical tokamak research and development gap, informed by the Spherical Tokamak Advanced Reactor (STAR) design study

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Monday Posters 1, Lobdell (Building W20 Room 208), June 23, 2025, 10:30 AM - 12:00 PM

In the new era of expanded private funding of fusion research and technology development there is substantial interest in low aspect ratio spherical tokamaks (STs) as compact fusion pilot plants (FPPs). The private company Tokamak Energy is pursuing this approach, as is the UK government through the STEP program. At PPPL public sector researchers are currently providing physics and engineering insights into the challenges and opportunities of a design of a Spherical Tokamak Advanced Reactor (STAR). Many of the physics projections for an ST FPP look favorable, but these are based on large projections in parameter space, mainly from results from the National Spherical Torus Experiment (NSTX). The process of designing the STAR device is telling us what research and development gaps need to be filled in the near future by experiments, including in the so-called Integrated Tokamak Exhaust and Performance (ITEP) gap. The upgrade to NSTX, NSTX-U, will be put into operation soon, and is targeting major performance increases to explore new physics regimes at low aspect ratio. The design process of STAR is informing the crucial elements of the NSTX-U mission, and the public-sector NSTX-U experiments will, in turn, be essential in informing the realization of future private-sector compact FPPs.

The operational design point of STAR at A = 2, R ~ 4m, and targeting 100-500 MWe net electric power was arrived at based in no small part on ST confinement scaling from NSTX. A major goal of the NSTX-U mission is to extend the confinement physics to lower collisionality through an upgrade of 2× toroidal field (0.5 ➔ 1T), 2× plasma current (1 ➔ 2MA), 5× longer pulse (1 ➔ 5s), at least 2× heating power (5 ➔ 10MW for 5s, up to 15MW NBI + 4MW RF for 1-2s) which will result in up to 10× higher $nT\tau E$ (~MJ plasmas). Pedestal height and width projections for STAR are also underlying the necessity of further experimental validation of new gyrokinetic critical pedestal models. The STAR design also assessed global stability and the possibility of non-inductive current drive, both of which will be explored further in NSTX-U at high beta and low collisionality. Alfven eigenmodes driven by fusion alphas were investigated as a concern for STAR and emphasized that NSTX-U will provide a unique regime for studying this physics, as fast ions from neutral beam injection can be super Alfvenic. Finally, power exhaust projections for STAR make clear the necessity of a heat exhaust solution and have motivated various designs of lithium divertors, including a so-called “lithium vapor cave”, that may be tested in NSTX-U, helping to close the ITEP gap.

Characterisation and Development of CVD Diamond-to-Metal Bonds for Microwave Components in Nuclear Fusion Systems

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Monday Posters 1, Lobdell (Building W20 Room 208), June 23, 2025, 10:30 AM - 12:00 PM

Advancements in nuclear fusion technology demand materials that can endure the extreme conditions present in high-power microwave components, particularly concerning the substantial power losses within waveguides [1]. These losses generate heat, leading to elevated temperatures and thermal gradients across materials. Chemical Vapour Deposition (CVD) diamond has emerged as a promising candidate for critical fusion reactor components due to its exceptional thermal conductivity and mechanical properties [2]. This study focuses on developing and characterising CVD diamond-to-metal bonds for microwave components serving as vacuum barriers in fusion systems. The research explores bonding CVD diamond to metals such as stainless steel (SS 316LN), copper (OFHC or CuCrZr alloys), and aluminium (6062).

Key analyses include pre- and post-cyclic fatigue testing to assess bond integrity in terms of vacuum sealing and thermal conductivity. Scanning and Transmission electron microscopy [3] and X-ray diffraction are employed to observe and characterise the behaviour of bonds under operational stresses. Preliminary findings aim to inform the development of reliable, high-performance components, advancing the integration of CVD diamond into nuclear fusion technologies. This work contributes to enhancing the durability and efficiency of systems operating under extreme conditions, supporting the progression of nuclear fusion as a sustainable energy solution.

[1] Kudryavtsev, I.V., Minakov, A.V. & Mityaev, A.E. (2019). The Influence of High-Power Microwave Signal Transmission on the Thermoelastic Condition of a Waveguide. J. Mach. Manuf. Reliab. 48, 306–313. https://doi.org/10.3103/S1052618819040101
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Development of a dynamic average-value model for superconducting coil power supplies in fusion pilot plant electrical systems

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Monday Posters 1, Lobdell (Building W20 Room 208), June 23, 2025, 10:30 AM - 12:00 PM

The electrical power system of a Magnetic Confinement Fusion (MCF) Fusion Pilot Plant (FPP) will differ significantly from those of conventional power stations. The requirements of personnel and plant protection, as well as grid connection criteria compliance, remain imperative. However, to these requirements are added the challenges of accommodating the pulsed demand of coil and Heating and Current Drive (H&CD) power supplies.

This work presents the development of a dynamic average-value model (AVM) representing a generic superconducting coil power electronic-based power converter. The model provides grid connection and coil power interfaces, as well as control signals for interlocking, DC-bus charge control, and output steering. The model also permits configuration of DC-bus voltage, control parameters and gains, and operational current and voltage limits.

The power supply model developed is demonstrated within a plant-level FPP pulsed electrical power system case study, a component of Tokamak Energy's FPP project, supported by the US DOE Milestone-Based Fusion Development Program. The power system case study includes multiple instances of the power supply model driving the superconducting poloidal field coils of the tokamak. The behaviour of the FPP pulsed power system during the phases of plasma ramp-up, flat-top, and ramp-down are demonstrated, with the mutual inductive coupling of coils and plasma current represented.

Advances of the High Flux Test Module (HFTM) towards a robust irradiation device for Fusion-First-Wall-Level high dpa at IFMIF-DONES

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Monday Posters 1, Lobdell (Building W20 Room 208), June 23, 2025, 10:30 AM - 12:00 PM

IFMIF-DONES is a material research facility under construction. It contains a deuteron particle accelerator and a lithium target to generate high energy neutrons at around 14 MeV. The target is enclosed in a test cell for radiation containment. Test modules are mounted into the test cell for irradiation campaigns to research material behavior relevant for future fusion plants, like DEMO.

An important test module positioned directly behind the backplate of the target is the High Flux Test Module (HFTM), which is currently being developed at Karlsruher Institut für Technologie (KIT), among others, as a part of the EuroFusion Work Package for Early Neutron Source (WPENS). The HFTM's objective is to allow high level irradiation (up to 50dpa) of small specimen test technology (SSTT) at well-defined temperature conditions. After an irradiation period of several month up to 2.5 years microstructural analysis and mechanical testing shall be performed. The HFTM and its instrumentation and control system shall allow to record as accurately as possible the temperature, neutron flux and neutron energy spectrum experienced by the specimens. The design of the HFTM aims to utilize as much of the available neutron flux as possible for sample irradiation by avoiding parasitic volumes and bulk materials. The HFTM's vessel walls, rigs and capsules in the beam footprint are therefore designed to be as thin as possible. Cooling is provided by miniature cooling channels in order to maximize volume of specimens in the beam footprint.

Pipes and Cables plugs (PCP) are located in the upper outer area of the test cell. The electrical signals, power supply respectively the helium coolant are fed and discharged via cable bridges respectively pipe bridges. These bridges are n-shaped and have connectors on the HFTM and on the PCP-side. The bridges allow for the compensation of manufacturing tolerances between the test cell and the HFTM and will be mounted with the help of robotic arms.

Previous design iterations included a "monolithic" container, i.e., it was planned to cut the cooling channels into the "container" (which contains the capsules containing the specimens) using Electric Discharge Machining. Our studies have shown that such a design is not feasible. We therefore present a composite design where the miniature cooling channels are formed between rig walls. The feasibility of manufacturing of similar thin-walled rigs has already been proven at KIT.

We have defined four load cases consisting of pressure and temperature profiles that allow us to demonstrate that the HFTM design iterations approach compliance with the RCC-MRx under all potential scenarios. We have carried out structural-mechanical Finite Element Analysis (FEA) for the entire HFTM vessel. Special care was given to accurately resolve stresses in the thin-walled (1,2 mm) parts of the container. We have proven that the entire HFTM is in compliance with the RCC-MRx for load case #2 (constant maximum pressure and temperature) and that our design brings us close to reaching compliance with the RCC-MRx for the load case #4 (pressure difference between compartments due to valve closure).

An Integrated, Systems and Stakeholder Centric Approach to Delivering LIBRTI, a Flexible Tritium Breeder Blanket Test Facility

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Access to data from relevant test facilities is essential to derisk future power plant breeder blanket technologies and guide their development. To address this need, a Systems Engineering approach was adopted in delivering the UKAEA LIBRTI (Lithium Breeding Tritium Innovation) facility. Scheduled to begin commissioning in 2027, LIBRTI is a pioneering flexible blanket irradiation test bed with 14 MeV Deuterium-Tritium (DT) neutrons. Its goal is to demonstrate controlled and repeatable tritium breeding at relevant temperatures on engineering scale blanket mock-ups. LIBRTI will support blanket concepts developed by the fusion community, with proposed breeder materials including solid ceramics, molten salt, and liquid metal. The user-designed mock-ups will be tested in front of the neutron source and connected to auxiliary services. The challenge lies in anticipating the needs driven by future potential users, whilst also recognising and satisfying other stakeholders, including the UK government, regulators, and suppliers. We propose a framework and methodology that ensures cohesive alignment between stakeholder needs and facility design to maximise the benefits to the wider fusion community.

Initially, the stakeholder needs on experimental and scientific objectives were elicited and prioritised. Techniques including the Ishikawa diagram were used to qualitatively identify the experimental parameters. Where necessary, assumptions were made based on a literature review of past tritium breeder experiments and proposed blanket concepts for fusion power plants. These, alongside legislative requirements, were fed into a single set of stakeholder needs. To encourage holistic life cycle thinking, a Concept of Operations (ConOps) was developed, looking at integrated scenarios of experiment commissioning, operation, removal, and decommissioning. The Product Breakdown Structure (PBS) was established to provide a framework for the project data. Deriving from the stakeholder needs, the Model-Based Systems Engineering (MBSE) method was used to perform a functional analysis. Functions and performance measures specific to each breeder concept were identified, forming a flexible operating envelope that can accommodate different experiments. These were formalised into textual system requirements and iterated with the MBSE model. System interfaces were identified between the facility and the mock-ups using Interface Control Documents (ICD). Throughout this process, the project conducted interdisciplinary reviews to validate the requirements and ensure that they address the stakeholder objectives.

As a result, scientific and engineering requirements were captured in a standardised method, recording the rationale and ensuring traceability to create a shared understanding across disciplines. These requirements guided sub-system optioneering towards defining the physical architecture, for instance aiding a multi-criteria decision-making process for a sparge gas system that enables sampling for dynamic tritium measurement. The ConOps provided a basis for HAZOP studies to identify experiment safety hazards and develop safety requirements. This set of traceable information ensures the project is robust, with a change management process established to evaluate the impact of future changes. Our findings recommend this approach for defining the problem space for future test facilities that face similar challenges.

Availability of the EU-DEMO Electron Cyclotron Heating Transmission Line and Launcher systems by Fault Tree and Reliability Block Diagram analyses

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The EU-DEMO reactor design is progressing through conceptual stages and reliability studies are performed on its systems to uncover any potential defect, weakness and imperfection that could be present. One of the paramount systems of EU-DEMO is the Electron Cyclotron (EC) Heating system. The EC system has the function to provide heating to the plasma during ramp-up and ramp-down sequence but also for other tasks such as Bulk Heating (BH), Neoclassical Tearing Modes (NTM) stabilization and Radiative Instability (RI) control.

As a first step of the reliability study, Failure Mode and Effect Analysis (FMEA) has been done to identify events that could impact the intended function of the system. For each component, failure modes, consequences caused by failures, and mitigation actions have been investigated. The individual failure events were categorized based on the Unavailability Conditions (UCs) they could cause.

Building on these findings, the next step of the study is a Reliability, Availability, Maintainability, and Inspectability (RAMI) analysis using Fault Tree (FT) and Reliability Block Diagram (RBD) models. The systems under interest for this second objective are the Transmission Line (TL) and Launcher section of the ECH. The FTs were done to calculate the mean time to first failure (MTTFF) for the equipment, considering all potential failure modes of its components. Meanwhile, RBDs were employed to estimate the availability of the integrated systems. Results are presented in terms of inherent availability of the TLs and Launcher systems integration over 20 years of operation. Additionally, parametric studies were conducted to assess the influence of the assumptions on the availability estimates.

Challenges of the DTT ECH Launchers and options under study

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The Electron Cyclotron Heating (ECH) launchers for the DTT facility are currently being designed. The aim is to inject power in different plasma locations with high flexibility to allow plasma current ramp-up and ramp-down, central heating with tailored deposition profiles and Neoclassical Tearing Modes (NTM) stabilization. Two antennas are developed, one for the equatorial port with six beam lines and one for the upper port with two beam lines, with similar front-steerable mirror modules. Each beam-line module is composed of a cooled corrugated waveguide launching a beam towards a fixed shaping mirror (M1) and then to the plasma-facing plane mirror (M2) movable around two axes, both cooled to sustain the high heat load due to the ohmic losses of the incident 1 MW beam and the plasma radiation. Thermal loads also include plasma and microwave stray radiation. Main structural loads on the mirrors and their supports, including the launcher supporting baseplate, are forces and torques due to the induced EM currents, during normal operation and disruptions, interacting with the high magnetic field in which the mirrors and supports are immersed. Thermal and structural challenges, in particular on the M1 and M2 mirrors, have opposite solutions that require delicate trade-off: optimal cooling requires high-thermal conductivity materials as copper alloys while low-electrically-conductive materials are needed for having lower induced currents during disruptions. Solutions under study for the mirror cooling for minimal deformations and stresses and for drive mechanisms are presented. The possible reduction of induced currents during disruptions using different materials, supporting structures and layouts is discussed.

Advanced Variance Reduction Applied to Plasma Source Term Variability Study of Poloidal Field Coil Energy Deposition

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At Commonwealth Fusion Systems (CFS), we are developing the SPARC fusion device, a high-field, compact tokamak designed to achieve net energy gain to demonstrate commercial viability of fusion energy. SPARC employs advanced high-temperature superconducting (HTS) magnets that enable stronger magnetic fields to maintain the plasma. These HTS magnets must be cooled to low cryogenic temperatures to function as designed.

The CFS Nuclear Engineering team performs detailed energy deposition calculations to assess neutronic heating to HTS magnets. A fine, detailed mesh of the Poloidal Field (PF) coils is used to understand the relative distribution of heating throughout the coils. For this radiation transport calculation, Monte Carlo N-Particle (MCNP) software coupled with advanced variance reduction techniques is used to evaluate the impact of different plasma source terms on PF coil heating.

The energy deposition calculation is performed using a Python Application Programming Interface (API) that enables an automated workflow to combine all aspects of the MCNP model for execution. Due to the extremely fine mesh in the PF coils, a significant number of particles are required to achieve acceptable statistics in the coil regions. To reduce the amount of compute time required to reach the necessary number of particles, variation reduction techniques are implemented. A Silver Fir Software (SFSW)-developed tool, Cottonwood, is used to generate weight-windows for the unstructured mesh model. The motivating tally for the weight-windows is set to all PF magnets to capture the impact of the changing plasma across different regions of the device.

The plasma source terms used in this calculation model a distribution of neutrons across a plasma region. The change in fraction of neutrons emitted from each plasma region is expected to change the energy deposition profile of the PF coils. Heating to the PF coils is calculated using energy deposition tallies (F6 tally) within MCNP. The integral heating for each PF coil is compared across the different plasma cases. Utilizing a fine mesh with variance reduction techniques enabled us to accurately capture small changes in heating as a result of different plasma profiles.

Multiphysics Open Source Workflow for Conceptual Design of Inertial Fusion Reactors

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This work presents an open-source workflow for the conceptual design of inertial fusion reactors, integrating neutronics, thermomechanics, and fluid dynamics through a consistent and automated approach. Neutronics simulations are conducted using the Fusion ENergy Integrated multiphysi-X (FENIX) code, which combines OpenMC via the Cardinal framework for neutron transport calculations and MOOSE for finite element thermomechanical analysis. A custom Python package is used to generate parametric reactor geometries, providing both constructive solid geometry (CSG) models for efficient sensitivity analyses and detailed CAD models using CadQuery. The CAD geometries are subsequently converted to DAGMC to enable detailed neutron transport simulations while ensuring geometrical consistency between the neutronics and thermomechanical workflows.

The neutronics analysis provides key metrics, including Tritium Breeding Ratio (TBR), displacement per atom (DPA), and heating deposition. The heating deposition results are then integrated into the thermomechanical workflow within FENIX, where thermal and structural analyses are performed to evaluate the reactor's response to operational conditions. These analyses are validated against results obtained using commercial tools such as Abaqus and open-source alternatives like Code Aster.

Thermal-hydraulics simulations are performed using a custom system-level code to analyze the performance of the coolant system, while computational fluid dynamics (CFD) simulations are used to study the behavior and stabilization of lithium jets, which serve as a protective layer for the reactor chamber. This workflow provides a comprehensive methodology for the integrated analysis of inertial fusion reactors, with a focus on establishing consistent processes and supporting future iterations of the conceptual design.

2D MHD Solver for Coupled Plasma and Liquid Metal Dynamics

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Liquid metal breeding blankets have several advantages over solid breeders, including simplified tritium extraction, dual functionality as a coolant, and elimination of mechanical stability concerns. However, liquid metal blankets are highly conductive and will be free to flow within channels, meaning they will directly interact with plasma dynamics via electromagnetic effects. In the process of blanket concept development and pilot plant design, there is a growing need for simple models that can effectively capture the interplay between the plasma and liquid breeding blanket. A 2D implementation of MUG [1], a component of the Open FUSION Toolkit (OFT) [2], is under development to solve the visco-resistive (in)compressible MHD with high geometric fidelity, which will be applied to predict liquid metal flows as a result of evolving magnetic fields. Once implemented and verified, MUG will be coupled with TokaMaker, a 2D MHD time-dependent plasma equilibrium solver already available in OFT [3]. This tool will be leveraged to study the impact of plasma disruptions on the blanket fluid, as well as if and how the blanket response alters the subsequent plasma dynamics.

[1] C. Hansen et. al., 2015 Phys. Plasmas 22 042505
[2] https://github.com/OpenFUSIONToolkit/OpenFUSIONToolkit
[3] C. Hansen et. al. 2024 Comput. Phys. Commun. 298 109111

Pellet Injection-based Model Predictive Control of the Density Profile in Tokamaks by Leveraging Deep Reinforcement Learning

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Monday Posters 1, Lobdell (Building W20 Room 208), June 23, 2025, 10:30 AM - 12:00 PM

Effective regulation of the plasma-density profile can play a critical role in achieving magnetohydrodynamic stability, sustaining optimal fusion conditions, and maintaining confinement in tokamaks. Traditional methods, such as gas puffing, face substantial time delays, especially in large-scale reactors like ITER. Moreover, gas penetration into the core may be limited by pedestal conditions in high-confinement operations. As an alternative, pellet injection offers faster response times with the capability of deeper penetration into the plasma core but presents at the same time significant challenges due to its discrete, on/off nature and complex control timing. This work introduces a novel model predictive control (MPC) framework enhanced by deep reinforcement learning to address these challenges. The proposed MPC scheme employs a linear control law capable of managing discrete control inputs and handling the nonlinear plasma dynamics by leveraging deep reinforcement learning. The plasma density profile evolution model is discretized in space by using the finite difference method (FDM), enabling the application of MPC. The dynamic pellet-injection process is modeled in discrete-time to accurately simulate fueling effects. Reinforcement learning (RL) optimizes control policies amid the inherent variability of pellet injection dynamics. Utilizing a deep Q-network (DQN) to handle the discrete control actions, the RL component adapts online to changes, improving control accuracy and convergence. Simulation studies validate the effectiveness of the proposed control strategy, demonstrating its capability to maintain the desired plasma density profile with high accuracy and reduced control effort.

Development of a Near Real-Time PF Coil Temperature Prediction System Using Deep Learning

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In this paper, we propose a near real-time superconducting coil temperature prediction system using deep learning. The PF coil is an essential component of the tokamak device, and KSTAR's PF coil must be cooled to 4.5 K to maintain superconductivity. However, due to AC loss caused by changes in coil current, the temperature of the PF coils fluctuates. To protect the magnet system, predicting coil temperature is important as abrupt temperature increases can occur during abnormal events, such as control failures. We developed a temperature prediction system based on a recurrent transformer model. The model takes previous temperature data and predict multiple temperature points at once. For near real-time applications, our system asynchronously collects input data from EPICS PV and performs deep learning-based predictions concurrently. To validate the system, we trained the deep learning model on KSTAR experimental data and tested it with experimental data. Our system demonstrated effective near real-time forecasting of PF coil temperatures.

Electromagnetic Analysis of Global ITER Electron Cyclotron Emission Model

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Monday Posters 1, Lobdell (Building W20 Room 208), June 23, 2025, 10:30 AM - 12:00 PM

The ITER Electron Cyclotron Emission (ECE) diagnostic system is located at Design Shield Mode 2 (DSM2), Equatorial Port 9 (EP9) to measure electron temperature profile and electron temperature fluctuations, and also assess non-thermal electron distributions via the oblique view. Therefore, ECE has both Radial and Oblique Views with two Hot Sources for calibration and two couples of mirrors for two different optical views. This ECE diagnostic system shall be exposed to significant thermal power due to unabsorbed electron cyclotron heating power in the plasma. It shall also receive the large electromagnetic (EM) loads up to 100MN/m3 force density due to the eddy currents on the copper mirrors generated by the short 16ms transient plasma disruption.

The simplified EM analysis model using magnetic field (B) data and flux variations (dB/dt) method based on the worst case of plasma disruption MD_DW_EXP16MS_CATIII has been developed to calculate the force and moments for each ECE component and support structure for EP9 DSM2 Bay2 and Bay3. The B and dB/dt method is using the constant B and dB/dt assumed for each local bay space, but the real B and dB/dt are basically varied along the radial direction of each bay. The complex global model with the local ECE components and support structure using Maxwell transient will be developed to do EM analysis to compare the results of B and dB/dt model. The volumetric EM force density of whole ECE components and support structure can be used for the subsequential structural analysis. Combined with the thermal and nuclear loads, seismic and initial loads, the ECE integration analysis can be finalized for the operation case. The detailed EM analysis results of Global ITER model with ECE components and support structures will be presented.

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A high-level overview of progress on the power exhaust system of Tokamak Energy's FPP

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Tokamak Energy is designing a Spherical Tokamak Fusion Pilot Plant (FPP) for integrated test and validations of technologies, systems and processes required for commercial fusion energy deployment. The FPP, which is targeting start of operations by 2035, will consist of an operationally-relevant fusion environment. By exploiting the inherent plasma physics benefits of the Spherical Tokamak, the FPP will demonstrate scalable net power in a fully-integrated system. Tokamak Energy and its FPP design efforts are supported by the U.S. Department of Energy's Milestone-Based Fusion Development Program.

Power exhaust is a major challenge in scaling from present-day devices to a power plant. A viable power exhaust solution must include plasma-facing components (PFCs) capable of withstanding the FPP environment (long service life, neutron irradiation, high heat and particle flux, electromagnetic loads), while effectively coupling heat to the power generation system. An edge plasma scenario which is compatible with both the PFC surface and the burning core plasma must be developed – this places challenging constraints on the magnet, matter injection and vacuum systems.

We will introduce the workflows and tools under development by Tokamak Energy for exploring the edge plasma operating space. We will also present outcomes to date in selection of edge magnetic topology and our corresponding strategy for heat and particle flux mitigation, as well as technology selection of PFCs including coolants, materials and heat removal method.

Error Analysis of OpenMC SPINS: Simulated Plasma Input for Neutron Source

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Monday Posters 1, Lobdell (Building W20 Room 208), June 23, 2025, 10:30 AM - 12:00 PM

In this contribution, we provide a detailed error analysis for the newly developed “Simulated Plasma Input for Neutron Source” (SPINS) library for OpenMC through comparisons with existing plasma source libraries in OpenMC. SPINS was created to allow flexible input of both simulated and experimental plasma profiles to generate fixed source neutron profiles in OpenMC. The toolkit implements 1D and 2D ion density and temperature profile input to determine volumetric neutron emissivity of the plasma using the Saddler-Van Belle formula. The library integrates the source function to sample neutron source locations proportional to the source strength, minimizing the introduction of sampling error. The method results in a source for which the sampled density of source sites is proportional to the actual density of the neutron emission rate in the plasma. While the volume integration enables a flexible input of simulated and experimental plasma profiles, it is accompanied by a discretization error which results from having a discretized source. Integration of sampled source sites and comparison to the original 1D profiles is performed to assess the numerical accuracy and effectiveness of the new presented method.

Activation analyses in support of the integration of the ITER Diagnostic Equatorial Ports

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ITER Diagnostic Equatorial Port Plugs (EPPs) are located in the Vacuum Vessel equatorial ports. Port Plugs are faced the plasma; they are directly irradiated by neutrons produced by the fusion reactions in plasma. Neutron irradiation results in heat release in port plug components and activates those. This work presents activation analyses in support of the integration activity of the ITER Diagnostic Equatorial Port Plugs (EPPs).
Three-dimensional neutronic analyses were performed in the past to calculate nuclear loads on the components of the equatorial port plugs #2, #8 and #12, housing different diagnostics systems. Calculations were performed with D1SUNED 3.1.4 rev.2 tool based on MCNP Monte Carlo code, using detailed representations of the ports integrated in the proper sectors from ITER reference MCNP model. In these analyses the neutron flux energy spectra were calculated in the port plugs and in the subcomponents, using a 175 VITAMIN-J energy group representation. These spectra have been used as an input for the activation analyses carried out with FISPACT II inventory code, employing TENDL-2017 nuclear data libraries and considering the SA2 ITER reference operational scenario. Specific activity, decay heat and contact dose rate of the port plugs components have been calculated after the machine shutdown for different cooling times up to 104 years. Results of the activation analyses at different cooling times will be presented in the paper along with a comparison between the ports highlighting major differences and similarities.

Design of a beam emission spectroscopy at W7-X

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Analyzing density fluctuations provides significant insights into turbulence and transport processes in fusion plasmas. In addition to the existing diagnostics for measuring density fluctuations at Wendelstein 7-X (W7-X), a new beam emission spectroscopy (BES) is envisaged for W7-X operation phase 2.4 starting mid 2026. The principle of this diagnostic is based on the observation of the light emission of a beam of neutral particles in a plasma, which are excited by collisions with plasma ions and electrons. The intensity of the light emission is proportional to the local plasma density. Compared to existing diagnostics, this method enables a significantly higher spatial and temporal resolution of density fluctuations. To achieve a high radial resolution the sightline must be tangential to the local magnetic field and as collinear to the beam as possible to provide a sufficient Doppler shift of the beam emission in order to allow a spectral discrimination of the beam emission from the background light. The latter can only be achieved for W7-X close to the edge of the plasma resulting in high radiative heat loads.

For this purpose, an optical system was developed that includes an immersion tube with a retractable, cooled mirror. By means of a retractable lens system in the immersion tube and a fiber bundle, an optimal image of the light emissions of the neutral particle beam is transmitted to a camera for analysis. This paper describes the concept of the beam emission spectroscopy system, its structure, its specific problems and the technical solutions for implementation at W7-X.

DIII-D Fiberoptic Bolometer Improvements

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Monday Posters 1, Lobdell (Building W20 Room 208), June 23, 2025, 10:30 AM - 12:00 PM

The Fiberoptic Bolometer, FOB has been optimized and installed on DIII-D. The FOB is an optical device that measures electromagnetic radiation in a vacuum environment. It utilizes a Fabry-Perot interferometer to convert incident radiation into temperature variations; these temperature variations are measured by discerning the shift in the reflection spectrum of the silicon cavity. The novelty of this design is that it is immune to electromagnetic interference because it doesn't use any electronics. Instead, optical signals are used to measure miniscule thermal variations. A previous iteration of this design was tested on DIII-D; several improvements have been identified and implemented. These improvements reduce the undesired black body radiation that was troubling the previous design.
The details of the design improvements are presented along with the computation of the harmful radiation mitigation. The FOB was designed to be Fusion Pilot Plant relevant because it utilizes all metal vacuum seals, is adjustable to various geometries, utilizes only vacuum compatible materials, and work is ongoing to prevent optical fiber darkening using germanium doped fiberoptics. The FOB's purpose would be to diagnose the FPP's plasma during the commissioning phase; more work will be needed to deploy this during the D-T phase to prevent activation. The FOB is fabricated primarily from stainless steel, copper and aluminum and utilizes mostly off the shelf technology. The base of the design is a vacuum vessel that bolts onto a DN200 conflat port of DIII-D. This vacuum vessel also supports a DN200 conflat aft flange, which plumbs in the eighty fiberoptic cables. The cables are connectorized using ferrule connector, angle polished (FC/APC) technology to minimize attenuation. A copper block is used to house the FOB's sensors. Its' secondary purpose is to act as a heat sink for the sensors to thermally equilibrate the bolometer array . The product development team has successfully installed and calibrated the system on DIII-D.

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Development of CADIS Capabilities in OpenMC using Adjoint Random Ray

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Measurements of critical performance metrics for fusion power plants such as fusion power, gain, and ion temperature and density rely on an extremely well-calibrated and characterized set of neutron diagnostics such as activation foils, fission chambers, and scintillator-based neutron cameras. The in-situ calibration procedure entails a lengthy series of measurements using a known standard of a neutron emitter such as Cf-252, and can be expensive to conduct frequently. Simulated calibrations using Monte Carlo particle transport can be used to determine the strategy and schedule of an in-situ calibration scheme, and assess sensitivities of detector responses to changes in the system geometry. Yet simulation-based calibration procedures are significantly limited by the degree of geometric fidelity of the model system and the computational cost of adequately sampling detector regions. This work explores the use of The Random Ray Method (TRRM) to facilitate Consistent Adjoint-Driven Importance Sampling (CADIS) as a new capability for local variance reduction on CAD-based tokamak models within OpenMC. The stochastic nature of TRRM in OpenMC is uniquely positioned to calculate adjoint neutron fluxes in complex fusion geometries where flux anisotropy is ubiquitous. From the adjoint flux, we generate weight windows and a corresponding source-biasing routine via the CADIS method over a single geometric representation. In this way, analyst time is not spent modeling the system in a deterministic code to generate the adjoint solution before transferring the corresponding weight window structure to a stochastic code to calculate the forward (neutron flux) solution, and interpolation between meshes does not take place. In this work, the TRRM-CADIS workflow is demonstrated on example problems relevant to fusion systems including both deep shielding and streaming gaps, and its performance relative to conventional discrete ordinates-Monte Carlo methods is assessed. As statistical uncertainties are decreased for a given investment of computational resources, this method can improve confidence in the calibration of detectors.

Research on Quality Management Evaluation Method of Power Supply System Installation Project for CSMC Test Platform of CRAFT

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To achieve nuclear fusion power generation, China is embarking on the construction of a key infrastructure - the Comprehensive Research Facility for Fusion Technology (CRAFT) project. The project aims to promote the early research and development of China Fusion Engineering Demo Reactor (CFEDR), focusing on the research and development and verification of the main engine and key components. The Central Solenoid Model Coil (CSMC) test platform is a sub-project of the CRAFT project. The management and evaluation of the power system installation project is key to ensuring the stable operation of the CRAFT power system, and is crucial to the successful construction and later safe operation of the superconducting magnet research system.

This article takes the installation quality evaluation of the CSMC magnet power supply system installation project as the research object, and uses the AHP-analytic hierarchy process to systematically evaluate the installation quality of the CSMC power system installation project. The installation quality is analyzed from the aspects of personnel factors, equipment and tool factors, construction material factors, and construction environment factors. Based on the evaluation results, suggestions are given for improving the quality of the installation project.

Design and implementation of fast transient overvoltage protection systems for SPIDER Power Supplies

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SPIDER is the full-scale prototype of the ITER Neutral Beam Injector ion source, operating since June 2018 at the ITER Neutral Beam Test Facility in Padova, Italy. The SPIDER ion source shares the same requirements as the ITER NBI ion sources in terms of RF power, operational pressure, geometry, and extraction voltage. A high-voltage power supply, known as the Acceleration Grid Power Supply (AGPS), provides the nominal acceleration voltage of -96 kV to the negative ion beam. Another set of power supplies, called the Ion Source and Extraction Grid Power Supply (ISEPS), is responsible for plasma generation, magnetic filtering, polarization of beam source components, and negative ion extraction up to 12 kV DC.

A distinctive feature of the SPIDER power supply system is its ability to handle frequent breakdowns during operation without stopping it, occurring both on the extraction stage and, more critically, on the acceleration stage, which result in load short circuits for the associated power supplies. Therefore, the extraction and acceleration power supplies were designed with fast switch-off time, minimized energy stored in the output filters, and output overvoltage suppressors. However, breakdowns on the acceleration stage generate fast electrical transients with high current and voltage peaks that propagate throughout the entire power supply system, impacting not only the AGPS but also ISEPS and other connected components, beyond the limits assumed during the design phase of each power supply.

SPIDER makes use of a large and interconnected electrical system, whose integration is not straightforward, since it depends also on stray parameters that were difficult to quantify during the design phase. While these parameters are typically negligible in industrial power systems, they become significant in SPIDER due to its complex system and to the presence of high-frequency transients characterized by hundreds of kHz and amplitudes of tens of kV.

This paper presents the analysis of the breakdown transients in SPIDER and consequent development of two specific protections to mitigate the overvoltage in critical positions of the power supply system: one on the AC medium-voltage grid supplying ISEPS and another at the output of the generator providing the extraction voltage to the beam source. Designing these protections posed significant challenges, including the need to limit fast transient voltage peaks caused by acceleration stage breakdown far exceeding the original design specifications of components connected to the extraction stage, the requirement for long operational lifetime with high reliability (managing up to 200 breakdowns per hour), and preventing damage to the power supply system in case of protection failure.

We conducted extensive tests on a set of prototypes, until they achieved sufficient maturity for installation in the plant.

The finalized protection system was then implemented and successfully tested in ISEPS under nominal conditions, with breakdowns up to the full acceleration voltage, ensuring readiness for SPIDER caesium operation in 2024.

Columbia Tritium Extraction Experiment (CTEX) for Molten-Salt Liquid Breeding Blankets

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A new testbed for tritium extraction is being developed at Columbia University. This experiment will demonstrate the feasibility of using molten salts in a vacuum-sieve-tray (VST) extraction system and examine the scalability to larger experiments and future pilot plants.

One scientific goal will be to experimentally validate the extractability of hydrogenic species from molten salts in a VST. The behavior and physical properties (permeability, diffusivity, and solubility) of the hydrogenated molten salts will also be documented in VST-relevant conditions. This will help refine computational models to predict and measure the extraction efficiencies of hydrogenic species from hydrogenated molten salt droplets. FLiNaK will be used as a proxy for FLiBe and other comparable breeding blanket-relevant salts. Another area of interest will be to validate the proposed internal convective mechanism within oscillating droplets, which should allow for higher extraction efficiencies and highlight the benefit of using a VST setup.

The experiment will be designed to incorporate modular diagnostics to study a variety of phenomena. A combination of pressure, temperature, and RGA sensors will be utilized to characterize the extraction efficiencies, and a high-speed camera will be used to capture the droplet oscillations. Additionally, different geometrical parameters, such as nozzle diameter and chamber height, will be studied to create a database to determine optimal conditions for maximizing hydrogen extraction. COMSOL and other numerical simulations have been used to baseline operational parameters. This experiment is designed as a batched operation rather than a closed-flow loop, allowing for easy swapping and iterating of components. Future diagnostics can be incorporated to study material corrosion, hydrogen permeability in steel, MHD effects in molten salt, and laser-induced spectroscopy for real-time fluid flow analysis.

This small-scale interface of physics, engineering, and material science will allow students from various fields to gain experience in the fusion technology ecosystem, engaging both graduate and undergraduate students.

This poster will highlight the current status of the project, which will include the computational models used to determine operational parameters and the engineering design of the experiment (vacuum systems, gas systems, heating methods, controls, P&ID, geometries, and diagnostics).

Development of an SRAM neutron detector for pulsed fusion experiments

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Radiation is known to cause single event upsets in semiconductor devices, which leads to data errors in SRAM memory (bit flips). While usually undesirable, this effect also can be used as a mechanism to detect radiation, in particular neutrons. Previous research shows that SRAM memory is capable of detecting neutrons without permanent damage to the device, and that devices from different processes and/or manufacturers vary in properties such as neutron interaction energy threshold and overall cross section.

In the work described here, an SRAM bit flip neutron detector was assembled, programmed and tested with a variety of sources. Two SRAM types with identical chip package and functionality, from Renesas and ISSI, were tested with a Cf-252 source, DD and DT neutron generators, and high yield neutron pulses from the Godiva (fission) and ZEUS (DT fusion) facilities. Additional tests were performed with decapsulated chips and an alpha source. We observe that the number of bit flips increases linearly with the neutron fluence. The ISSI SRAM is physically a smaller die, but shows 2-100 times more bit flips per neutron than the Renesas SRAM, depending on the neutron source. The bit flips are distributed randomly across memory addresses, except for groups of three close neighbors observed in the Renesas SRAM. No damage to the SRAM has been observed, even in the intense EMI and high instantaneous flux of the ZEUS pulsed neutron fields. With a suitable calibration for bit flips per incident neutrons, the detector can be used for quantitative measurements, and there is potential to use the ISSI vs Renesas type to differentiate between DT and DD neutrons. With a cost of <$10 per memory chip with 4 million bits, the design is highly scalable into large arrays for higher sensitivity. The chips are also conducive to rapid replacement in the case of operational damage. On Helion Energy's Polaris fusion generator, the SRAM detector will be fielded as a relative yield neutron fluence detector with regular calibrations against indium and zirconium neutron activation foils.

Cryogenic Application of Austenitic Stainless Steels for Superconducting Magnet Structures in SPARC

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Structural steels with high-strength and high-toughness at cryogenic temperatures are key to enabling high-field, compact, cost-effective tokamaks such as SPARC. The high magnetic fields generated by high-temperature superconducting (HTS) magnets scale linearly with stress in the magnets and their supporting structures requiring the use of special, high-strength steels. CFS has identified, characterized, and qualified austenitic stainless steels to facilitate SPARC magnet construction. A custom composition of 316LN(H) has been selected as the primary cryogenic structural material due to a combination of favorable mechanical and non-magnetic properties as well as commercial availability at scale. CFS, in collaboration with its partners, has produced 316LN(H) components for SPARC in multiple sizes and form-factors. The mechanical performance has been characterized and meets or exceeds the structural requirements needed for SPARC operation. Not only do these materials need to have exceptional properties, they also must maintain high performance when welded. CFS has identified manganese modified 18Cr-16Ni-5Mo-0.16N L filler metal coupled with a GTAW or GMAW process that can produce high performing welds that can withstand SPARC loading. The characterization of 316LN(H) performance at cryogenic temperatures and challenges in scaling steel structures to grid-scale fusion devices will be presented.

Design of a Diamond-Based In-Vessel Soft X-Ray Camera for the SPARC Tokamak

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Soft x-ray tomography is an important diagnostic on modern fusion devices, however the silicon diode photodetectors that are used for in-vessel soft x-ray detection in many tokamaks are sensitive to neutron damage, making them unsuitable for burning plasma devices like SPARC. This sensitivity can be mitigated by placing the detectors far from the plasma, limiting their field of view and making tomographic reconstruction impossible; or by using a neutron-hard in-vessel photodetector. One promising detector technology is single-crystal diamond photodiode, which has been demonstrated to have excellent tolerance against 14 MeV neutrons [1] and tested in high temperature environments [2]. These diamond photodiodes have been demonstrated to work in a tokamak environment in the form of single detectors [3]. Presented here is the design of a camera made from an array of these diamonds. SPARC will employ several of these cameras in a single poloidal plane and their overlapping fields of view will enable tomographic inversion of the emissivity profile [4]. The high temperature, high neutron flux, and disruption forces that will be experienced in SPARC present a unique design challenge that is addressed here using simulation and planned physical experiment.

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Design of a workflow for ITER synthetic Infrared diagnostics

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The visible and InfraRed (IR) Wide-Angle Viewing System (WAVS) in ITER plays a critical role in real-time machine protection and plasma control, while contributing to plasma-wall interaction studies. As was shown in[1][2], the modeling of the WAVS IR synthetic images is important to predict the diagnostic performance in the ITER all-metallic environment.
The goal of this work is the development of synthetic diagnostics to generate realistic IR camera images for ITER, supporting the design of First Wall (FW) Heat Load Control functions. The present analysis is therefore focused on the FW region, monitored by the Equatorial port viewing system (EWAVS).
The FW heat load controller for ITER will be based on the evaluation of the thermal loads in real time with a combination of surface temperature measurements from IR cameras and reconstruction of the plasma heat flux impacting the Plasma Facing Components (PFCs)[3].
In this work we present the design of an innovative workflow for realistic estimation of heat fluxes and First Wall Panels (FWPs) temperature evolution as monitored by the IR cameras. The model allows the identification of a set of Regions of Interest (ROI) to be specifically monitored by the WAVS, defining the real-time inputs to the plasma control system.
Plasma generated heat loads are reconstructed using the magnetic field line tracing tool SMITER, which accurately maps the distribution of heat fluxes on PFCs. These reconstructions provide boundary conditions for thermal models that simulate the surface temperature response using a 2D transient model for heat diffusion in the FWP depth. The presented workflow is used to generate synthetic images representing WAVS IR camera measurements under several ITER operational scenarios, providing a performance assessment of the diagnostics and enhancing their reliability.
The reduced models used for FW surface temperature evolution combine a realistic dynamics and are also studied for their potential real-time application. Validation of these models is conducted using data from CEA experimental setups[2] and smaller tokamak experiments designed to assess high-heat-flux conditions and IR camera performance.
Preliminary results discussed here demonstrate the accuracy and reliability of the models, establishing them as robust tools for improving the safety of the ITER operation. To improve the fidelity of the model, as future work, the reflections will be taken into account using the ray-tracing code CHERAB. The integration of surface temperature predictions and plasma heat flux reconstructions enables rapid machine protection actions, while addressing the complex reflection mechanisms significantly enhances diagnostic reliability. The synthetic images also provide a comprehensive diagnostic assessment, serving as a critical tool to evaluate the performance of the WAVS IR systems under realistic operational conditions. The methodologies and findings presented here can be extended to other tokamaks, contributing to the design of advanced diagnostics and machine control frameworks.

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An M3D-C1 Implementation of the SPARC Runaway Electron Mitigation Coil

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Runaway electrons (REs) generated during disruption events in tokamaks can carry mega-Ampere level currents, potentially causing localized melting and vaporization damage to plasma-facing components. Should the runaway electron risk predicted for future high-current devices be realized, mitigating runaway electrons will be of paramount importance for the efficient operation of SPARC. We evaluate the effectiveness of a Runaway Electron Mitigation Coil (REMC) – a passively driven magnetic coil with an n = 1 geometry – to prevent the growth of runaway electrons in SPARC. The REMC achieves this by seeding magnetohydrodynamic (MHD) instabilities during the current quench, generating field-line stochasticity that deconfines REs faster than they are generated. Using 2-D and 3-D MHD simulations in M3D-C1, coupled with externally imposed REMC fields, we perform high-fidelity simulations of RE and magnetohydrodynamic activity during disruption events on SPARC. M3D-C1 uses a fluid RE model, which is evolved self-consistently with the bulk MHD plasma, allowing us to evaluate the efficacy of field-line stochasticity introduced by the REMC in preventing RE growth. Future work will also explore incorporating the REMC directly into the M3D-C1 wall model, enabling fully self-consistent simulations of RE mitigation via the REMC in M3D-C1.

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Design of active disturbance rejection control strategy for resonant magnetic perturbations coils power supply based on fractional order ESO

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Abstract: The dynamic current output performance of the resonant magnetic perturbations coils power supply in a Tokamak device is limited by the integral term of the traditional proportional-integral (PI) control strategy, resulting in suboptimal performance. Additionally, the disturbance suppression capability of the PI control strategy proves inadequate in the complex electromagnetic environment of the vacuum chamber. To address these challenges, this paper established a fractional-order mathematical model of the power supply system, accounting for the fractional-order characteristics of inductors and capacitors. Based on this model, a fractional-order extended state observer (FO-ESO) is designed to estimate disturbances in real time, treating them as an additional state of the system. Furthermore, a transition process is incorporated to achieve fast current output without overshoot. Simulation and experimental results demonstrate that, compared to the traditional PI control strategy and the standard active disturbance rejection control (ADRC) strategy, the fractional-order active disturbance rejection control (FO-ADRC) strategy significantly improves the dynamic characteristics of high-current output in the power system. Specifically, it achieves faster response speed and shorter output delay time. Moreover, the proposed strategy exhibits enhanced robustness and anti-interference performance in the complex electromagnetic environment of the vacuum chamber.

Key words: resonant magnetic perturbations coils power supply; active disturbance rejection control; fractional order characteristics; extended state observer

Commissioning of the power supplies and coils of the SMART tokamak

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The SMall Aspect Ratio Tokamak (SMART) is a spherical tokamak (ST) that offers unique capabilities for studying the potential of negative triangularity. It has been designed, constructed, and is currently being operated by the Plasma Science Fusion Technology Laboratory (PSFT) at the University of Seville. SMART has a total of 21 coils, organized into 7 independent circuits and driven by modular power supplies (PS). The construction of the PS has been made with switching converter technology based on IGBTs and supercapacitors [1]. PS provide the current profile for standard copper coils: central solenoid, toroidal field system (12 coils), 3 pairs of poloidal field (PF) coils, and 2 independent PF coils [2].

This study presents the commissioning and validation process carried out on the PS and coils at SMART. The actual operating parameters of the PS were measured during one second pulses. The maximum rated current and slope were obtained, while the series impedance of the coils, the output current ripple, induced noise, and the temperature rise were also measured. A comparison has been made between the theoretical values in [1] and the experimental results, providing insight into the performance of PS and areas for improvement.

The set-point currents are stored and programmed in the memory of the PS prior to activation. During plasma discharge, the current profiles of two independent PS are controlled in real time (RT) by a CompactPCI-Express PC, improving experimental flexibility and optimizing the control of vertical displacements. The hardware architecture of the RT control is described and a simplified assembly model is also presented to facilitate the upcoming integration of the MARTe2 framework [3]. Magnetic sensor measurements were taken to tune the variable time settings of the closed-loop RT control system [4]. The results obtained provide essential information on the requirements of the RT system for response time, signal integrity, and operational reliability.

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Automating neutronics and structural engineering for stellarator fusion pilot plants

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Engineering design of magnetic confinement fusion devices requires various calculations ensuring e.g. sufficient tritium breeding in the blanket (TBR), minimization of neutron heating to superconducting magnets, and building support structures for magnets. In particular, stellarator coils require careful design to ensure the design meets both engineering and physics requirements and constraints due to traditional modular coil shaping and requirements for proximity to the plasma. We have undertaken furthering the computational tools and workflows for these tasks. First, using the parametric CAD generator parastell, OpenMC neutronics simulations at the CAD-level are performed to calculate TBR and nuclear heating of magnets. Variance reduction techniques are required for good statistics for the local nuclear heating in the magnets, and need to be further developed for better targeting of the regions of interest. Additional engineering analysis workflows have been developed to optimize structural element placement between modular coils, using ANSYS structural optimization tools, in combination with finite-element analysis of the electromagnetic forces on the coils along with structural calculations of stress and strain. Placement of lateral structural supports are automated to minimize the displacement of the magnets below a nominal value, with future work to tie in directly with plasma equilibrium optimization tools.

High-fidelity Neutronics Analysis for LD-FIRST (Laser Driven Fusion Integration Research and Science Test Facility)

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Accurate predictions of neutron and gamma transport in inertial fusion energy systems are essential for assessing structural material damage, heat deposition, and tritium breeding. This work uses a workflow for high-fidelity simulations to examine the Laser Driven Fusion Integration Research and Science Test Facility (LD-FIRST) which is Lawrence Livermore National Laboratory's (LLNL) newest proposal for a full-scale community testbed for fusion energy concepts. The proposed workflow utilizes a high-resolution CAD-based model provided by LLNL, meshed for use in a neutron transport simulation, enabling the mapping of neutron damage, energy deposition, and tritium breeding with high accuracy. While previous analyses used one- or two-dimensional calculations to estimate the parameters of interest, this high-fidelity workflow facilitates fully three-dimensional simulations, offering detailed visualizations to identify critical regions of neutron damage, optimize tritium breeding locations, and refine energy deposition profiles. By applying this advanced neutronics modeling capability, this study provides insights into the performance and feasibility of LD-FIRST. The results will inform design optimizations, material selection, and shielding strategies necessary for the successful development of this test facility. Ultimately, this work supports LD-FIRST's role in advancing inertial fusion energy research by providing a robust framework for assessing neutron transport effects in next-generation fusion systems.

Progress on the Fusion REactor Design and Assessment (FREDA) SciDAC Project

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Future fusion pilot plants will operate in regimes far beyond current experience, and device design will rely on physics-based prediction and extrapolation. Many concepts will also rely on simulation to assess safety or to reveal and solve the complexities of system integration that may otherwise not be apparent in physical components or models developed in isolation. The Fusion REactor Design and Assessment (FREDA) SciDAC project is building a component-based, integrated plasma and engineering modeling framework and data structure to enable self-consistent, iterative optimization workflows for fusion reactor design. FREDA aims to provide a set of flexible workflows to support various stages of the design process using a multi-fidelity model hierarchy, ranging from the simple analytic descriptions to the highest fidelity, theory-based plasma and engineering modeling developed by the fusion and fission communities. Goals and initial progress on the project objectives will be presented. The plasma simulation backbone is IPS-FASTRAN with newly developed coupled Core/Edge Pedestal/Scrape-Off-Layer (CESOL) workflows. A major gap for plasma+engineering coupling is predicting heat and particle flux to and from the wall. Workflows have been developed to map the heat flux from the far-SOL region of the plasma up to the first wall and divertor using both lower fidelity and higher fidelity (BOUT++ and UEDGE) methods. Improvements have been made to automatically generate mesh for plasma boundary modeling codes to enable scans in geometry and physical constraints. FREDA incorporates the FERMI engineering modeling suite for self-consistent evaluation of the thermal shields, limiters, blanket, magnets, and other surrounding structures with predictions of temperatures, erosion, dpa, activation, tritium generation and transport, creep, corrosion, material degradation, etc. Parametric generation of 3D CAD enables rapid iteration of component geometry in response to plasma and loading specifications. FREDA parametrization tools are being developed and tested to a) generate parametrized CAD components from user defined modules called within the geometry generation, b) take an existing high fidelity CAD and defeature for lower fidelity/faster analysis, or c) take an existing CAD and create parametrization for optimization. Uncertainty quantification methods (DAKOTA) are being implemented to assess the importance of parameters and model uncertainties within the system analysis and enable design under uncertainty to reach desired device robustness and reliability.

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F4Enix, a new python library for the automation of neutronics workflows

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The computation of nuclear responses for fusion reactors is a complex and resource-intensive process. To give an order of magnitude of the problem, E-lite, which is an MCNP detailed model of the entire ITER tokamak, is composed by more than 400 thousand cells and the text output files produced by simulations performed on these models can easily occupy more than a few Gb of storage. Dealing with all this armed only with text editors becomes virtually impossible and calls for heavy automation to reduce human errors and speed up the neutronic analyses workflow. To address these kinds of challenges, a great number of scripts was produced during the years at Fusion For Energy (F4E). These scripts presented a wide range of maturity, verification level and documentation quality. For this reason, the neutronics team of F4E recently decided to collect all these codes, refactor them, generalize them, and aggregate them into a unique pip installable python package named F4Enix. This allowed to simplify maintenance, add and extensive documentation, allow synergies between the different parsers and add continuous integration and deployment features. The library is essentially an API that allows to parse a wide range of file types that are commonly used in neutronic analyses workflows. F4Enix is not the only python package developed with this scope, but it presents unique characteristics with respect to other established libraries such as PyNE. First, it is written in pure python, which helps with installation and integration with other packages. Second, it includes parser for many types of files that are not available in other packages. For the moment, F4Enix allows to parse and manipulate (with varying degree of completeness) the following file types: MCNP input, output, meshtal, mctal, rssa, eeout and wwinp files; FISPACT II legacy output (only to extract decay pathways); D1SUNED additional files such as meshinfo, irrad and react. To conclude, even if F4Enix started as an internal F4E project, it is quickly being introduced to the F4E suppliers and, hopefully, to the whole neutronics community.

A mesoscale phase-field model of intergranular liquid lithium corrosion of ferritic/martensitic steels

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Lithium-bearing liquid metals, such as liquid lithium (Li) and the eutectic lead-lithium alloy (Pb-Li), are being investigated as breeder materials for fusion reactors. Aside from their beneficial qualities, including no concern for radiation damage and superior thermal properties, they give rise to severe corrosive environments leading to weight loss and surface recession of the structural materials thus negatively impacting the structural integrity. Literature details Li penetrates structural materials via grain boundaries (GBs); intergranular corrosion (IGC), resulting in chemically altered regions surrounding the GBs in turn deteriorating mechanical properties such as hardness. Consequently, it is essential to be able to predict this phenomenon in order to understand the governing factors and in turn optimize the material and its microstructure to produce a corrosive resistant material to liquid Li. Phase-field modeling (PFM), a widely employed computational method to simulate corrosion as the underlying equations are built in a thermodynamic consistent manner, is utilized to simulate the IGC process. Through a diffuse interfacial region, PFM tracks the interface between two phases: metal and corrosive agent, implicitly using a phase-field order parameter thereby eliminating the need for restrictive interfacial boundary conditions. The diffuse interface, in turn, allows for the exploitation of experimentally determined physical properties that ultimately governs the intergranular penetration of Li based on diffusion-dependent kinetics. The resultant dissolution of alloying elements is consequentially tracked to evaluate the severity of corrosion. GBs and grains are distinguished via an additional order parameter, solved independently to the primary kinematic variables, thereby decreasing computational expense. Attributed to its high reactivity and immature employment, there remains little experimental data regarding the interfacial properties between liquid Li and structural materials, namely interfacial energy and the diffusivity of Li. Consequentially, focus is placed on microstructural properties, including GB energies and solid-state diffusivity of alloying element, to circumvent this obstacle. The model is validated against experimental measurements for a ferritic/martensitic steel specimen exposed to static liquid Li. Thereafter, the microstructural influence on the corrosion process was surveyed by altering grain properties, including grain size, GB density and GB thickness. It was found near-surface GB density plays a non-trivial contribution to the performance of the material. Yet, it was evident the grain size, which dictates the bulk GB density, ultimately governs the severity to IGC. Subsequently, the incorporation of saturation on the corrosion process shifted the apparent behavior of the material such that the microstructural influence on corrosion depth became distinct in turn dampening the correlation with weight loss. Moreover, the model has been extended to incorporate the kinetic effect of fluid flow on the corrosion process to better emulate the dynamic conditions of a breeder system. The model has been further extended to incorporate EBSD data, allowing for the impact of high- and low-angle GBs on the corrosion process, to emphasize the corrosion performance under higher fidelity conditions.

Virtual Integration Platform for Engineering and Remote Handling (VIPER)

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Maintaining and installing equipment in nuclear confined environments present significant challenges, demanding precise planning and advanced technologies to ensure worker safety and reduce radiation exposure. The Virtual Integration Platform for Engineering and Remote Handling (VIPER) offers an innovative framework to enhance the design, simulation, and operational efficiency of fusion energy projects. The platform's immersive simulation capabilities offer realistic virtual environments for validating complex engineering processes. These capabilities are crucial for high-stakes nuclear operations where precision and safety are critical.

VIPER focuses on four interconnected domains: digital continuity, scale one immersive simulation, nuclear applications, and Remote handling, leveraging state-of-the-art tools and methodologies. This abstract highlights VIPER's application in key fusion use cases, including WEST tokamak component assembly (e.g., Plasma Facing Units, ICRH, and TWA antennas), ITER Tokamak maintenance and decommissioning activities and DEMO remote maintenance.

In particular, the platform develops a stand-alone 4D viewer with integrated functionalities such as sequence management and integration of the temporal dimension. By automating coupling to model-based systems engineering (MBSE) principles and robust metadata management, the application guarantees data accuracy and consistency throughout the design and operation phases. In addition, the coupling with the XDE physics engine modules enables the integration of physical analyses of operational feasibility, accessibility and ergonomics, as well as collision detection and path-finding algorithms. Fully integrated into the engineering design process, it significantly reduces information access times and improves team communication.

VIPER's virtual environments also support the conceptualization and testing of fusion machine designs, fostering stakeholder collaboration and refining design iterations. The platform's evolution embraces emerging technologies including investigations with new standards for enhanced interoperability, real-time collaboration, and scalability. The integration of Artificial Intelligence technologies will further optimize processes accelerating decision-making and improving overall efficiency.

VIPER demonstrates transformative potential in fusion energy projects. Its unified platform for design, validation, and remote handling enhances projects collaboration. With a forward-looking roadmap, VIPER is poised to advance digital engineering methodologies, solidifying its role as a cornerstone for the next generation of fusion energy systems.

Keywords: Virtual Reality, digital continuity, immersive simulation, nuclear engineering, Remote Handling

Design and analysis of CFETR thermal shield

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Thermal Shield (TS) is a key component of Chinese Fusion Engineering Test Reactor (CFETR). The role of TS is to reduce radiation heat load transferring from the vacuum vessel (VV) and cryostat to the magnets operating at 4.5 K. The TS of CFETR shall be properly designed so that the radiation heat load is below the specified requirement for the magnets. There are 16 upper ports, 6 equatorial ports and 16 lower ports for the CFETR TS. For TS, it is of importance to ensure that it can maintain thermal loads. In this paper, a study is conducted on thermal analysis of TS. The total heat load and temperature distribution are obtained. And structural analyses (normal operation and accident condition) are also made to verify the strength of the TS structure under several typical conditions. Based on the finite model (FE) of TS, the design of TS can satisfy the design requirements after optimization.

Conceptual engineering analysis modeling of ECCD launcher design for JA DEMO

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The electron cyclotron current drive (ECCD) system design activities for JA-DEMO is on-going. The physical analysis of ECCD in JA-DEMO presented RF injection conditions and launcher design had started. Due to severe radiation condition, the remote steering system is planned to be applied in JA-DEMO launchers.

In this study, we will present a conceptual design of the upper port launcher for engineering analysis modeling. To avoid the harmonic absorption region of ECCD in JA-DEMO plasma, an angle of more than 60° in the toroidal direction is required for upper port injection, which is difficult to achieve by direct output from the end of the waveguide using the remote steering system. Therefore, we performed a conceptual design with a fixed front mirror that is angled in the toroidal direction in front of the remote steering system. In the design, we planned to use 25 rectangular waveguides with sides of 50 mm for remote steering, arranged in a 5x5 configuration, and to inject about 25 MW from a single port. The movable mirror mechanism for remote steering, which is installed behind the injection device, is installed outside the cryostat.

The structure of the launcher consists of waveguides surrounded by the shielding structure which is stainless steel with water piping for radiation shielding and cooling. The engineering model is developed for evaluation of the spatial radiation dose rate behind the port where the steering mirror system locates. In addition, maintenance planning considering access to the equipment in the device will be reported.

Adaptive Control of Plasma Density in Tokamaks Using Coordinated Gas Puffing and Pellet Injection

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Regulating plasma density at desired levels is crucial for achieving optimal performance in tokamaks. While gas puffing has proven effective for fueling current devices, its performance is expected to degrade in future reactor-grade tokamaks such as ITER due to increased device sizes and higher plasma densities. This will lead to longer transit times from gas valves to the plasma edge and limit gas penetration into the core. Furthermore, advanced plasma conditions will demand more precise and rapid control to ensure optimal and safe operation. To address these challenges, pellet injection has emerged as a promising solution, capable of delivering fuel directly into the plasma core. However, the practical implementation of pellet injection for density control remains largely unexplored in current tokamaks. The discrete nature of pellet injections, variations in pellet size and injection speed, and the travel time of the pellet in the tubes all pose significant performance challenges. To address these challenges, an adaptive control technique has been developed for the regulation of the line-averaged electron density. The proposed approach coordinates gas puffing and pellet injection while accounting for uncertainties in actuator responses and rapid shifts in plasma conditions. Simulation studies indicate that the adaptive strategy is well suited to regulate density around desired targets for a variety of operating scenarios, offering a robust path toward consistent fueling in next-generation tokamaks.

Decoupling the hardening mechanisms in irradiated materials

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Reduced Activation Ferritic/Martensitic (RAFM) steels are strong contenders for structural applications in fusion demonstration reactors, including the Spherical Tokamak for Energy Production (STEP). Compared with austenitic alloys, RAFM steels offer superior thermal conductivity, low thermal expansion, and high resistance to void swelling, rendering them particularly suitable for the severe conditions encountered in fusion environments. Nonetheless, irradiation-induced hardening, arising from a complex interplay of defects such as dislocation loops, bubbles, and precipitates, degrades their mechanical performance. Decoupling the individual contributions of these defects is therefore central to optimising material behaviour and ensuring the long-term reliability of fusion reactor components.

Results from two complementary irradiation studies will be presented. In the first, EFDA FeCr alloys (9–12% Cr), EFDA ultra-high-purity Fe, and T91 steels were exposed to 2 MeV protons at a fluence of 4.0×10^{17} protons/cm^2, generating displacement damage levels between 0.02 and 0.2 dpa. These conditions replicated previous 20 MeV Fe-ion studies, which demonstrated that doses as low as 8×10^{-5} dpa can cause significant hardening. Therefore, the use of 2 MeV protons facilitated the formation of dislocation loops, voids, and precipitates akin to those induced by 14.1 MeV fusion neutrons.

In the second study, the role of yttrium-titanium-oxygen (Y-Ti-O) nano-oxide precipitates in managing irradiation-induced defects was assessed by implanting 2 MeV protons into oxide dispersion-strengthened (ODS) and non-ODS variants (14YWT and 14WT) to roughly 0.02 dpa and 0.2 dpa. The key aim was to determine whether nano-oxide dispersions could mitigate defect accumulation caused by transmutation reactions in fusion environments, ultimately diminishing swelling and embrittlement.

Post-irradiation nanoindentation revealed clear contrasts in the hardness response of the examined alloys. In 14WT, a pronounced increase in hardness and modulus emerged at 18–19 µm below the irradiated surface, coinciding with the region of peak damage. This abrupt rise in mechanical properties points to the dominant influence of defect clustering and dislocation interactions in the non-ODS alloy. In comparison, 14YWT demonstrated a far more uniform hardness profile, suggesting that Y-Ti-O precipitates act as effective defect sinks by promoting vacancy–interstitial recombination and limiting dislocation loop and void formation. These findings reinforce the role of nano-oxide dispersions in enhancing the radiation tolerance of ODS steels by limiting defect accumulation and maintaining mechanical stability.

This collected data is being integrated into crystal plasticity models to examine how specific defect types influence irradiation hardening in ferritic/martensitic steels. Once refined, these models will capture critical failure mechanisms, such as plasticity and fracture, and streamline the selection process for future fusion reactor concepts.

Viability study of the use of fusion energy to tackle freshwater shortage

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The United Nations World Water Development Report 2024 published by UNESCO warned about the imminent risk of a global water crises, and the prospects of this exacerbating conflicts worldwide. According to this report, 2.2 billion people still live without access to safely managed drinking water, i.e. more than one quarter of the world's population. Between 2002 and 2021, droughts affected more than 1.4 billion people, and climate change is expected to accentuate this problem. Furthermore, freshwater demand has increased by almost 1% since the 1980s, and it is likely that it will continue to grow as populations urbanize. Some countries have palliated the demand for drinking water by using desalination plants powered by fossil sources. However, this results in a huge emission of greenhouse gases: the pollution in 2020 amounted to around 76 million tons of CO2 per year and is increasing year after year as more desalination plants powered by fossil fuels are opening [1]. Therefore, nuclear desalination emerged as a cleaner alternative, where the electricity produced in the nuclear power plant (NPP) is used to power the desalination plant, or the thermal energy generated is used for a thermal desalination process, in a dual-purpose mode. This is where fusion could play a crucial role.

As in any classical power-to-electricity conversion cycle, large low-grade heat losses are expected in fusion NPPs. Some possible methods to boost the efficiency of future fusion NPPs such as hydrogen production [2] and district heating [3] have been analysed. In this work, we explore the viability of using fusion energy to produce freshwater, by studying the integration of a fusion NPP with a Multistage flash desalination plant as the heat recovery technology. A set of desalination plant layouts and power cycles are modelled using the numerical solver EES [4]. The boundary conditions (heat sources and temperatures) were obtained [2] with the PROCESS code [5,6], taking as reference the spherical tokamak pilot plant presented by Menard et al [7]. A portfolio of Rankine water cycles and both helium and supercritical CO2 Brayton cycles are presented, with efficiencies in the range 0.3-0.5. The economic viability, which considers production and distribution costs (i.e., penalization in the electricity production and pumping consumption) is discussed.

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Coordinated Relay Protection Scheme for Integrating Future Spherical Tokamak Fusion Power Plant with Microgrid Concept

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With the rapid progress in fusion research worldwide, its integration with the electrical power sector is becoming a critical topic of discussion. Unlike the large-scale fusion power plants in Europe and the United States, the UKAEA's STEP programme focuses on commercializing relatively compact fusion power systems, aiming to export electrical power above 100 MWe to the national grid. The STEP Prototype Powerplant (SPP) adopts the microgrid concept of Distributed Energy Resources (DER) to achieve higher flexibility and economic efficiency. However, its unique operational characteristics present significant challenges to relay protection systems. The substantial variations in power flow magnitude and direction, caused by the unique operational mode of the system and the integration of large-scale DC energy storage devices, result in diverse fault characteristics that could potentially lead to relay maloperation or misoperation, posing a risk of irreversible damage to the fusion reactor. The integration of large-scale DC energy storage systems introduces rapid fluctuations in power flow and bidirectional energy transfer, which complicate fault detection and isolation due to non-linear transient behaviours and dynamic fault impedance. Presently, relay protection tailored to such systems has not been fully addressed.

This research proposes a novel coordinated relay protection scheme tailored to the unique operational characteristics of SPPs. The scheme includes adaptive fault detection algorithms, relay coordination strategies, and real-time fault isolation mechanisms. Based on official UKAEA data, the UK Grid Code for STEP, IEEE Std C37.105-2010, and IAEA Safety Standards No. SSG-34, the proposed approach considers a wide range of fault scenarios to ensure robustness and adaptability. By leveraging RTDS RSCAD software simulations, the study validates the scheme's feasibility through comprehensive modelling of power supply systems, including grid connections, on-site generators, and energy storage devices. The results demonstrate optimized power generation and storage control, improved fault analysis accuracy, and enhanced system reliability across diverse operating conditions. This work makes a significant contribution by addressing the gap in relay protection for fusion power plants and providing a practical solution that advances the integration of fusion energy with modern electrical grids. Supported by the STEP programme, this study lays a foundation for the commercial deployment of fusion power systems.

Development, Testing and Commissioning of 300 kVA T-NPC Inverter Power Supply for High Power Particle Accelerator

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This paper describes the developmental aspects and features of 300 kVA, 400 V, 400 Hz three-phase T-NPC IGBT based DC/AC Power Converter (Inverter) system as is being implemented with Testing and Commissioning results. High Accuracy (≤ 1%), Stability (≤ 1%), wide range of output voltage control (5%-100%), Fast transient response time (≤ 100 µsec), and Low THD (≤ 5%) are the requirements of the Power Supply System as it is integrated with un-controlled High Voltage AC/DC Power Converter (HV-Rectifier) which in turn supplies the stable DC Power to Particle Accelerator based application like Neutral Beam Injection (NBI) system in Fusion R&D program. The Inverter is connected to front-end 12 pulsed SCR based AC/DC Power Converter (Controlled-Rectifier) forming the DC-Link. The fine and course control of the output voltage is achieved by the Controlled-Rectifier and Inverter respectively as per the requirement of load.

The validity of design performance and configuration is established and presented in this paper under both steady-state and transient operating state.

Keywords: Inverter, Topology, Testing, Response Time, THD, High Voltage DC Power Supply

Concept and challenges of the DTT ECH transmission lines

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The Electron Cyclotron Heating (ECH) system of the DTT tokamak foresees at present an installed power of 32 gyrotrons (1 MW/170 GHz/100 s), located in a building ~100 m from the tokamak hall and connected to it with a bridge. Sources are grouped in four clusters, each one equipped with its own power supplies, Transmission Line (TL) and launchers. The TL is entirely based on the quasi-optical propagation of beams and the layout is conceived with initial and final sections with Single-Beam (SB) mirrors and a main intermediate section with Multi-Beam (MB) mirrors. The connection between the first SB section and the MB and between the last MB mirror and the final SB section is realized through beam combiner and splitter units, respectively. All the mirrors’ units are housed in evacuated pipes to avoid the risk of arcing. The performance of the TL has been evaluated in terms of spillover fraction and coupling with the fundamental TEM00 mode at the end of the TL both in the ideal case and in case of deformation of the optical surface of the mirrors, due to ohmic losses, and for reasonable misalignments. To limit the former, mirrors need to be actively cooled, and the cooling layout has to be designed to limit temperature gradients. Misalignments include deterministic sources, such as the effect of strong winds on the suspended corridor, and random sources for installation errors. Supporting structures of the mirrors and housing units will be designed to ease installation and to allow fine regulation after the installation. Two MB mirrors shall be also equipped with actuation systems to recover the alignment between successive pulses, if needed.

A digital twin framework to support SPARC boundary physics operations

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Digital Twins (DT) are computational objects designed to mirror real physical objects via a combination of real time diagnostic data with computer models. DTs are used across many industries to track the state of a component over the course of its operational lifetime, to predict component failure, and to improve component performance. Commonwealth Fusion Systems (CFS) used Digital Twin Prototypes (DTP) to design the ~25k SPARC plasma facing components by coupling plasma power exhaust models directly to the 3D engineering CAD and manufacturing tolerances. As PFCs are manufactured and installed, metrology data across multiple sources will be combined with CAD models to generate 3D as-built CAD models of the PFCs. Every SPARC pulse will employ 3D PFC state tracking for temperature, stress, and tungsten recrystallization on an interpulse timescale. Visualization of the 3D BDT output will be accomplished by a standard plotting routines or by a high speed graphics render engine.

Now, as CFS prepares for operations, this DT framework is being extended for use in the control room by boundary physics operators. This Boundary Digital Twin (BDT) will connect forward models across a range of physics and engineering domains. Plasma edge physics codes will provide radiation profiles and target heat and particle fluxes. These physics codes will be coupled to 3D engineering finite element models which provide temperature and stress distributions throughout the PFC in addition to recrystallization kinetic predictions. Synthetic diagnostic and instrumentation models will be developed for the BDT that can be compared to real experimental diagnostic data, and autonomous discrepancy identification algorithms will be capable of flagging mismatches between BDT diagnostic predictions and SPARC data and tuning the relevant physics models to generate good agreement. Machine learning models will be employed to accelerate the BDT forward models where needed, and data analytics across libraries of simulation and experimental data will identify trends and patterns. The BDT will enable the boundary operators to achieve an accelerated experimental schedule for SPARC while running the PFCs at maximum performance without compromising protection. The CFS BDT project has only recently begun, a roadmap has been drafted, and resources are being allocated across CFS and our affiliate collaborators. This work will summarize the work already completed for the PFCs using a BDT, will highlight the CFS roadmap for this project, and will provide a few example use cases for the framework.

Acknowledgement: Supported by Commonwealth Fusion Systems.

Crystal Plasticity Finite Element Modelling of Cyclic Deformation in OFHC Copper for Heat Sink Component under Low Cycle Fatigue

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Oxygen-Free High Conductivity (OFHC) copper has been proposed as a candidate material for the heat sink interlayer in fusion reactor divertor components such as DEMOnstration Fusion Power Plant (DEMO). However, gaps exist in their service life prediction, which remains insufficient to ensure reliable engineering design and safety for advanced nuclear fusion reactors. This work aims to address these gaps by conducting Low Cycle Fatigue (LCF) tests to evaluate the performance of components subjected to the cyclic loading inherent in tokamak operations. We present the results of LCF tests conducted on OFHC copper at room temperature and elevated temperature relevant to its operational environment in DEMO. The tests were performed at a total strain range of 0.6%-0.9% and a strain ratio of R= -1, providing insights into the estimated fatigue life and cyclic deformation behaviour. Fractographic analysis using scanning electron microscopy (SEM) was also performed to examine crack propagation and microstructural damage. The results revealed the cyclic softening behaviour of OFHC copper, a characteristic that has also been observed in Eurofer97 ferritic-martensitic steel. This work integrates a crystal plasticity finite-element model (CPFEM) incorporating a phenomenological hardening law. The model accounts for kinematic hardening to simulate the cyclic hardening/softening behaviour providing an understanding of the microstructural mechanisms at play during cyclic deformation. By bridging the gap between experimental observations and predictive modelling, the findings contribute to developing reliable design frameworks and safety standards for future fusion reactor components.

Kyoto Fusioneering America: Overview of Activities, Achievements and Prospects

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Kyoto Fusioneering Ltd. (KF) is a private fusion energy company headquartered in Tokyo, Japan. KF's mission is to serve the fusion industry by developing auxiliary thermal and fuel cycle systems, in addition to gyrotron plasma heating that provide the necessary technologies on the way to a fusion pilot plant (FPP). To address the technological challenges related to these systems, KF is developing the UNITY program which includes a thermal cycle test facility named Unique Integrated Testing Facility-1 (UNITY-1) housed at the Kyoto Research Center in Kyoto, Japan, a fuel cycle facility called UNITY-2 being built in Chalk River, Ontario, and two blanket loop facilities, one which uses FLiBe molten salt and the other pure Li, which are both located at Kyoto University. This work will show Kyoto Fusioneering America's (KFA's) major areas of ongoing research and how the research has served KF's broader goals of developing UNITY, initiating collaboration with other entities to engage them in the program, and accelerating the development of the first FPP.

KFA is based in Seattle, Washington and is the US subsidiary of KF. KFA is involved in a number of activities from foundational R&D to fusion commercialization, which includes fostering both technical and business relationships in North America that further KF's goals whilst supporting the growth of the U.S. fusion community. Many of these initiatives are DOE-led, and KFA has remained proactive at staying involved in this space. For example, one of the areas in which KFA has had successes is in the Innovation Network for Fusion Energy (INFUSE) program. KFA has won four INFUSE grants to date, working with SRNL, PPPL, INL, and ORNL. Respectively, these projects explore lithium compatibility for electrode materials, MHD modeling in flowing Li-Pb, investigation of tritium breeding ratios in Li-6/Li-7 after thermal neutron irradiation, and Li-Pb interactions with blanket materials with varying Li and Pb concentrations. These projects have allowed KFA to establish strong working relationships with public partners, and access technical expertise that has helped further both UNITY and the US fusion program.

KFA will stay engaged in the INFUSE program, as well as look to prospects in other initiatives such as the Fusion Innovation Research Engine (FIRE) and Private Facilities Research (PFR) programs, allowing the public sector to access KF's capabilities.

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Improvement of EAST density limit scaling formula based on symbolic regression

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Plasma density is a key parameter in fusion plasmas, directly affecting the realization of ignition conditions and the basic operation of future fusion reactors. Although the Greenwald density limit is widely used to predict the maximum achievable density of tokamak plasma, significant deviations from this limit have been observed in experimental data from the EAST device, revealing the limitations of the Greenwald density limit scaling. To address this issue, this study proposes a symbolic regression-based approach to derive an improved density limit scaling formula using density limit experimental data from the EAST tokamak.
Specifically, in the first step we use the LightGBM algorithm for feature selection to identify the plasma parameters that have the greatest impact on the density limit, including the scrape-off layer power (PSOL), the safety factor (q95), the plasma elongation (κ), and the minor radius a. These features are taken in reference with the key parameters mentioned in the literature [M. Giacomin et al., Physical Review Letters, 2022]. In this study, the MARFE event (multifaceted asymmetric radiation from the edge movement) is regarded as a precursor of the density limit, and the average density value at the moment of its start of movement is chosen as the density limit value. Secondly, a symbolic regression model based on genetic algorithm is used to optimize the parameters of the selected features, and the new density limit scalar formula is obtained by fitting in power law form .The results show that the new model fitting coefficient of determination R2 reaches about 0.70, which is much higher than the R2 fitted using Greenwald formula,which is only 0.17. In the derived scaling formula, the power of the scrape-off layer, the safety factor, and the elongation are found to be of high importance for density limit scaling. Since the value of a on the EAST device is basically stabilized around 0.45, its influence on the density limit scaling formula is relatively small. The derived formula is effectively captures the multivariate complex interaction effects that are not adequately considered by the traditional methods, which significantly improves the accuracy and physical consistency of the density limit prediction. This study demonstrates the feasibility of combining machine learning techniques with experimental data to derive the density limit scaling formula.

Conceptualization of a Multi-Purpose Testing Facility for Advanced Fluid Dynamics Investigations in Tokamak Divertor Modules

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A novel testing facility is being conceptualized to experimentally support the development of prototypical components for divertors in Tokamak fusion reactors. The facility will be conceived to operate in a wide range of thermal-hydraulic conditions, with the capability to reach high pressure (up to 150 bar) and high temperature (up to 250°C) conditions, enabling rigorous testing of the fluid dynamics and mechanical behaviour of components.

The primary focus of the facility will be the investigation of fluid dynamics, particularly around swirl flows and within the targets of Plasma Facing Units (PFUs). With the use of advanced diagnostic tools, including a four camera to a Tomographic Particle Image Velocimetry (Tomo-PIV) system, the facility will enable in-depth analysis of fluid flow behaviors. This will provide valuable insights into how fluids interact with complex geometries, helping to optimize the thermo-fluid dynamic performance of fusion reactor in-vessel components.

In addition, the facility will support experimental campaigns focused on vibration and fluid-structure interaction analysis. Flow-induced vibrations are critical factors that could potentially limit the service life of key in-vessel components in a nuclear fusion reactor, particularly those involved in the heat extraction system (e.g., the first wall, divertor, etc.). By employing high-sensitivity measurement systems, both contact (accelerometers) and non-contact (Laser Doppler Vibrometer, LDV), the facility will enable the measurement and subsequent analysis of the vibrational spectrum and the dynamic interaction between fluids and solid structures (e.g., pipe-fluid-swirl) under various operating conditions, as well as the vibrational characterization of the components through Experimental and Operational Modal Analysis (EMA and OMA). These insights will enhance understanding of the mechanical stability and dynamic response of the components, contributing to their reliability and longevity.

With its combination of cutting-edge diagnostic tools and experimental versatility, the facility will play a pivotal role in advancing research in fluid dynamics and structural mechanics for fusion technologies. The integration of these experimental techniques into a unified platform will foster interdisciplinary collaboration and enhance the efficiency of research efforts.

Design and Optimization of the RFX-mod2 Vacuum System: challenges, performance analysis and solutions

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The RFX-mod2 design introduces several novel features, such as the in-vessel passive stabilizing shell (PSS) supporting the first wall (FW) and an array of in-vessel sensors, including magnetic, thermal, and electrostatic diagnostics, along with extensive cabling housed within the vacuum chamber. Additionally, the presence of two Viton O-rings, collectively spanning over 20 meters, introduces unique challenges. These new elements significantly alter the vacuum system's behavior compared to its predecessor, RFX-mod, due to the emergence of increased outgassing rates from in-vessel components that were previously absent.

This work focuses on the evaluation and optimization of the vacuum system required for RFX-mod2. The analysis considers the performance of the existing RFX-mod pumping system, which includes 10 turbomolecular pumps, each with a nominal pumping speed of 630 l/s for H_2. While the high vacuum stage is maintained without modifications, the medium and low vacuum stages are undergoing a overall refurbishment. These revisions are aimed at improving reliability and compatibility with the upgraded system, even though they do not contribute directly to an increase in the overall pumping speed being the volume of the vacuum vessel substantially unchanged.

The study encompasses detailed calculations to assess the achievable vacuum levels and identifies potential bottlenecks introduced by the new design features. These include evaluating the effects of outgassing from the copper shell (covered by alumina coating for electrical insultation purpose), the graphite first wall, and the large Viton O-rings. The findings are used to refine the vacuum system's design, ensuring it meets the stringent operational requirements necessary for the successful implementation of RFX-mod2.

By addressing the challenges posed by the upgraded machine's configuration, this work provides a comprehensive framework for designing and implementing a vacuum system capable of supporting the enhanced performance goals of the RFX-mod2 experiment.

Investigation of the optimal operational space of NBTF solid-state RF amplifiers

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The Neutral Beam Test Facility (NBTF) plays a crucial role in advancing plasma heating and current drive technologies for ITER. It aims to develop and test the high-performance 1 MeV neutral beam injector (MITICA) and its negative ion source (SPIDER). To achieve the required negative ion density, each experiment within NBTF employs four RF generators that supply as many couples of plasma drivers, for a total rated power of 800 kW at 1 MHz.
The RF power supplies are currently being upgraded from tetrode-based oscillators to solid-state amplifiers and are at the final procurement stage. This transition aims to address the instabilities observed during experimentation with the oscillators, which prevented the achievement of nominal power. Solid-state technology is expected to improve the operation of individual RF generators; however, other side effects exist related to the simultaneous operation of the four units. Previous observations indicate that the power delivery of RF generators to the load is influenced by mutual coupling effects between the circuits they power. These effects cause the impedance seen by individual generators to oscillate or deviate from optimal conditions, depending on frequency synchronization and phase relationships among them. To enhance the operational efficiency and reliability of the new solid-state amplifiers, we conducted an extensive analysis to identify and address potential optimization areas. In particular, this study aims to expand on our previous considerations on coupling-induced impedance deviations, to optimize power delivery to the load. This emphasizes the importance of precise frequency and phase alignment to guarantee efficient performance. Given the complexity of achieving this with the existing fixed-elements and narrowband matching network, we explore various matching strategies to enable effective isofrequency operation.

Design of a Symmetric Traveling Wave Antenna for Fast Ion Production on DD Tokamaks

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Future burning plasmas will have appreciable populations of energetic fusion-born alpha particles. Here, a system is proposed to experimentally explore the effects of these fast populations on existing D-D tokamaks via selective RF heating of neutral beam particles through high harmonic fast waves with deuterium cyclotron harmonics of 5 to 9. The proposed system is a high field side symmetric center-fed traveling wave array antenna. Novel antenna features include the center feeding, high field side launch, and additional straps dedicated to image current cancelation for reduced impurity production.
The physics scenario was scoped on a selected DIII-D discharge for high single-pass damping and good selectivity for ion damping over electron damping using the ray-tracing/Fokker-Planck codes GENRAY [1] and CQL3D [2]. The antenna design workflow developed includes a custom optimization tool built using a cost function that designs for launched power spectrum, image current cancelation, and low reflection coefficient. The tool uses a combination of finite element method (FEM) analysis software including COMSOL [3] for vacuum optimization, and the Petra-M FEM multi-physics framework [4] for cold plasma optimization, as well as Python RF network analysis packages. We show that the workflow presented here can produce a traveling wave array antenna with a desired power spectrum, reflection coefficient, and reduced impurity production potential via the cancellation of image currents, and that such an antenna placed on the high field side produces an attractive system to produce fast ions in the correct region of parameter space, warranting future more detailed studies. Recommendations are given for how best to take advantage of the traveling wave arrays' ability to passively impedance match using the optimization framework developed here.

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Impact of 14 MeV-neutron exposure on SiC varistors for nuclear fusion applications

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Recent nuclear fusion systems are based on superconducting coils that must be rapidly discharged in case of faults, such as quenching, to protect both the coils themselves and the entire system [1]. This operation is standardly performed by inserting a linear resistor in the coil circuit [2]. The discharge time obtained with resistors is inversely proportional to the peak voltage produced across the coils, so in many cases it is impossible to identify a good trade-off between the two contrasting requirements.
In the last years, the use of varistors as an alternative to linear resistors for fast coil discharge has been under investigation. Varistors are bi-directional semiconductors characterized by a non-linear relationship between voltage and current. In particular, silicon-carbide (SiC) varistors have demonstrated high-energy absorption capabilities [3] and the possibility to operate over a wide temperature range [4]. Due to these properties, they are being considered for installation in next-generation fusion machines to enable the emergency discharge of superconducting magnets, also offering further operational advantages. For example, in the Divertor Tokamak Test (DTT) facility the use of varistors allowed a compromise between peak voltage and hot-spot temperature in the design of toroidal coil fast discharge units, and an increased operating range for the central solenoid switching network units [5].
In addition, the use of varistors appears particularly attractive for protection of in-vessel components as stabilizing plates and coil feedthroughs [6]. In this context, research is ongoing to develop new production processes capable of meeting the required specifications.
Although several experimental characterizations of SiC varistors have been conducted, there is still a lack of studies concerning the effects of neutron irradiation on these devices. This work aims to address this lack by presenting a study on the electrical performance of SiC varistors after exposure to different doses of 14 MeV-neutron irradiation at Frascati Neutron Generator (FNG) facility. Multiple SiC samples were electrically characterized both before and after neutron irradiation, also using a test facility able to replicate discharge conditions. Samples that successfully pass the high-radiation characterization tests will be used to produce varistor prototypes that meet the requirements for nuclear fusion applications. These prototypes could be installed in existing fusion machines and further optimized for future fusion experiments, such as DEMO.

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Engineering and Integration of Microwave Diagnostics at SPARC: Electron Cyclotron Emission and Edge Scanning Reflectometry Systems

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Microwave diagnostics at SPARC include the Electron Cyclotron Emission (ECE) and Edge Scanning Reflectometry (ESRL) systems, which are currently in their final design stages. These diagnostics face significant engineering challenges, such as constrained space, high thermal and electromagnetic loads, lengthy transmission lines (approximately 20 meters), and radiation shielding requirements for laboratory components. Additionally, the limited accessibility of the tokamak area during operation highlights the critical need for the development of in-situ calibration techniques for these systems.

The ECE system is a key diagnostic designed to measure the core electron temperature profile. Operating at probing frequencies in the 245–455 GHz range, it will provide measurements in the first harmonic O-mode at 12T magnetic fields and the second harmonic X-mode at 8T. The system offers high spatial resolution with 24 radial channels, enabling at least 15 plasma measurements for magnetic fields ranging from 11.5T to 12.2T, and will achieve a time resolution of 0.1 ms or better. The system incorporates millimeter-wave hardware, optical components (mirrors), a beamline, and vacuum windows, all of which must perform reliably in SPARC's demanding operational environment.

The ESRL system will provide edge electron density profiles using both O-mode and X-mode measurements across probing frequencies in the 18–90 GHz range. It is designed to measure electron densities from $3x10^{18}/m^3$ to $4x10^{20}/m^3$, corresponding to 0.004 - 0.5 of the Greenwald density limit for the maximum plasma current (8.7 MA) operation. The ESRL system employs a bi-static arrangement of horns and waveguides, vacuum windows, and a backend processing system, overcoming similar constraints encountered by the ECE system.

Both ECE and ESRL systems share a midplane port and laboratory space, emphasizing the need for efficient integration and interface optimization. Notably, the ECE beamline in the ex-vessel section doubles as a support structure for ESRL waveguides, optimizing physical space utilization. These systems advance the state-of-the-art in microwave fusion diagnostics, such as novel calibration techniques for reflectometry and advanced hardware designs, especially for vacuum windows.

This work was supported by Commonwealth Fusion Systems.

CREEP-FATIGUE LIFETIME ASSESSMENT OF HELIUM COOLED PEBBLE BLANKET

BREEDING BLANKET

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The urgency of addressing the climate crisis has accelerated advancements in fusion energy research. A pivotal challenge is the development of breeding blankets, a key in-vessel component essential for tritium breeding and thermal power generation, which are critical for demonstrating fusion's commercial viability. Within the European DEMO program, the Helium-Cooled Pebble Bed breeding blanket (HCPB BB) is one of two designs under consideration. Extensive studies, including neutronics, hydraulic, thermal-structural, and tritium breeding capability analyses, have been conducted to evaluate its performance. Previous thermal-structural studies of the HCPB BB primarily employed the nuclear design code RCC-MRx for design verification against immediate plastic collapse (IPC), immediate plastic instability, immediate plastic flow localization (IPFL) and progressive deformation (PD) rules. As the HCPB BB is a plasma-facing component subjected to extreme operating conditions, with high temperatures ranging between 300–550 °C, one of its primary failure modes is creep-fatigue. Therefore, the current study aims to extend these efforts by utilizing RCC-MRx creep-fatigue rules to develop methodologies for creep-fatigue lifetime assessment. Thermo-structural analysis was conducted based on previous studies, and the resulting stress field was linearized at the regions of interests in structural components: front wall, pressure tube and back support structures. . A MATLAB-based program, incorporating RCC-MRx creep-fatigue rules and P91 alloy steel material properties, was developed to evaluate the creep-fatigue life of these paths. The results reveal significant variations in creep-fatigue life, ranging from approximately 22 million cycles in the back support structures to as few as 356 cycles in the pressure tube, mainly driven by difference in the temperatures used for the material properties and the stresses. Furthermore, the creep fatigue analysis was repeated using material properties at average (400 °C) and maximum temperatures (550 °C) instead of linearized path temperatures. The results showed a reduction in variations in creep-fatigue life, with values ranging from approximately 650 cycles to 14 cycles, highlighting the conservatism in using simplified material operation temperature limits to inform breeding blanket design. This is due to the regions limiting lifetime being affected by a combination of temperature, stress fields, radiation damage, and helium accumulation. This study unlocks the potential to combine robust breeding blanket design, assessment, and alloy development, enabling the rapid identification of lifetime limiting structural regions of interest, operational conditions, and key properties that need improvement.

DTT PFCs Cages and Supports and TFCs Casing: Design Analysis and Optimization

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DTT is an experimental nuclear fusion facility located in Frascati's ENEA Research Center. The aim of this device is studying how to deal with the huge thermal and parcticle fluxes in the region of the divertor, in a way to make possible to achieve the nuclear fusion as a reliable and stable power source. The magnetic system of the device includes 3 different kinds of magnets: 18 Toroidal Field Coils (TFC), 6 Poloidal Field Coils (PFC) and a stack of 6 modules for the Central Solenoid (CS). All the magnets are built by superconductors operating at a temperature of 4.5K.
The casing structures of the Toroidal Field Coils (TFCs) and the support structures of the Poloidal Field Coils (PFCs) are essential to ensure the mechanical stability and optimal performance of the Divertor Tokamak Test (DTT) facility's magnetic system. These structures must be carefully engineered to withstand the extreme electromagnetic forces and thermal stresses encountered during operation. This paper provides an in-depth analysis of the technical requirements, design considerations and engineering strategies employed in the development and fabrication of these components. Key aspects of the design process include the selection of structural configurations capable of enduring cryogenic temperatures (4.5K) and high magnetic fields, while also addressing industrial constraints such as cost-effectiveness and simplicity of construction.

Experiment Control of the Novatron N1 Mirror Machine

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The Novatron [1] is a novel axisymmetric magnetic mirror-cusp concept. The magnetic field configuration produces favorable curvature at the outer plasma surface that should be inherently stable against interchange instabilities that have limited previous magnetic mirror devices. The first iteration of the Novatron concept has recently become a reality as the N1 machine has been built and first ionization tests with hydrogen plasma by ECRH heating have been performed at the Alfvén Laboratory at the Royal Institute of Technology (KTH) in Stockholm, Sweden.

This poster presents an overview of the different technical subsystems of the N1 machine as well as the architecture of the experiment control and data acquisition software NovaXC (Novatron Experiment Controller) and the experiment and data management backend NovaDB (Novatron Database).

During plasma experiments NovaXC coordinates all subsystems of the N1 machine including machine control (vacuum, magnets and fuelling), RF heating, diagnostics timing and data acquisition. The application is implemented in LabVIEW using the open DQMH framework which provides the basis for a modular, scalable and maintainable architecture. It supports a modern team-focused development process including source-code control, unit testing and continuous integration. The application runs on a PXI chassis which also holds the main data acquisition and signal generation instruments; however, external instruments and devices can be supported. Each instrument is encapsulated in an application independent code module. The experiment control specific functionality is implemented in subsystem code modules which follow a defined state sequence for experiment shot execution. Subsystems define their configuration parameters as JSON Schema, which allows validation and simplifies the interface to NovaDB. Further key features include an intuitive operator interface, automated experiment sequence execution, and comprehensive data logging using the hierarchical data format (hdf5). The architecture's modular design allows for rapid integration of new subsystems and changes to experimental parameters while maintaining operational stability. Performance validation has demonstrated reliable control of experiment parameters and successful automation of complex pulse sequences with dozens of subsystems during multiple experimental campaigns.

NovaDB is our experiment and data management system. It is used to store contextual and searchable data, in a SQL database, together with experiment data in a HSDS backend. The core of NovaDB is its C# Core web service that implements a REST API on top of a domain-driven design model that allows storing, fetching, filtering and searching data. A web app written in HoloViz Panel exposes an overview of the experiment data in a modern web UI and allows administering the configurations and recipes used during shot execution. A python client package allows querying the actual datasets in python scripts for deeper data analysis.

NovaXC and NovaDB have been successfully deployed in the ongoing experiments with the N1 machine with potential applications in other plasma experiments requiring precise temporal control and synchronization.

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Implementation of a gyro-orbit model in M3D-C1 code for simulating MHD-induced energetic particle transport in the SPARC tokamak

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Monday Posters 2, Lobdell (Building W20 Room 208), June 23, 2025, 2:00 PM - 3:30 PM

Energetic particle (EP) transport is a critical challenge in the burning plasma scenarios of the SPARC tokamak. Understanding the interaction between EPs and various MHD events in SPARC could provide valuable insights for improving confinement and performance. To explore these dynamics, the SPARC primary reference discharge (PRD) case was used to investigate potential MHD instabilities that may arise during SPARC operation. Linear M3D-C1 simulations revealed a dominant kink mode with toroidal mode number n=1. The corresponding eigenmode structure appeared at multiple q flux surfaces, which could get close to the edge plasma near the last closed flux surface (LCFS). Investigating whether such fluctuations lead to significant EP transport is relevant to SPARC performance. To address this, a kinetic (gyro-orbit) particle pusher was implemented in the M3D-C1 code, enabling the tracing of EP trajectories in the presence of MHD instabilities with SPARC equilibria. Initial investigations demonstrate how energetic particles can be passively transported by the saturated kink modes in nonlinear 3D simulations. These findings provide a foundation for self-consistent modeling of EP dynamics coupled with MHD instabilities using M3D-C1. Furthermore, the results can be validated against various EP diagnostics that are planned for installation on SPARC in the near future.

DIII-D Tokamak Remote Vessel Inspection

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The DIII-D tokamak is one of the most highly diagnosed tokamaks in the world. However, there is not a dedicated system to view the vessel under vacuum in the visible light spectrum. To inspect the inside of the vessel, the machine must be vented. Rudimentary cameras on a stick have been used to look through an unpopulated port but this is limited to a small number of ports that remain unpopulated. Inspection of the Lower Hybrid Current Drive launcher was recently attempted, and unsuccessful, using the pole method due to the lack of an available port near the launcher. The highest fidelity inspection is human entry into the vessel which requires several days of work on either end of the inspection. Human entry is possible on DIII-D due to the D-D only operations but infeasible for a fusion power plant or experimental D-T tokamak. DIII-D would benefit from a system that allows freedom to move about the inside vessel without the need for human entry.

A system for inserting a first-person view vehicle into the vessel will be presented. Challenges include selecting a vehicle form factor that allows for entry via a small flange (avoids an overhead crane to remove the flange), insertion /retrieval, illumination, mobility and diverse perspectives / points of view.

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Development of the remote handling connector for DTT divertor diagnostic system

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At the DTT plant in Frascati (Italy), a new solution was implemented to connect the large number of diagnostics on the diverter test cassettes to ensure high system availability.

The different types of electrical and optical cables used by the diagnostic sensors installed on the cassette must be able to be connected to the diagnostic room for the facility to function properly. Four test cassettes are available on which the diagnostics are concentrated; they are designed to reduce maintenance time.
The connector must also be easily removable and acts as a bridge between the cassette and the hall. It operates in a small space under environmental conditions of high vacuum, baking temperatures, radiation and electromagnetic forces without compromising reliability and functionality. The preliminary design of the diagnostics connector compatible with remote handling was strongly influenced by the need to create a system whose full functionality is always recoverable in the event of failure of one of its parts. The design, although at a preliminary stage, required an iterative process of requirements gathering, mechanical design, FEM analysis and RAMI analysis.
The mock-up is under construction and will be tested at DTT's Remote Handling facility, which is also under construction. The test results will be important for the final version of the connector used on the cassettes.

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Design and Evaluation of Inertially-Cooled Tungsten Plasma-Facing Components for the DIII-D Tokamak

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The DIII-D program is embarking on a project to replace all of its graphite plasma-facing surfaces with tungsten to address key core-edge integration challenges for fusion energy. A staged implementation is proposed, where high heat flux surfaces are solid tungsten while lower heat flux (including recessed) surfaces consist on W coatings on a compatible metallic substrate. For the first stage of installation in 2027, the DIII-D lower divertor and the three poloidal limiters at midplane will be replaced with ~5mm thick tungsten panels on copper mounting blocks ("pedestals"). These panels will be "fish-scaled" for a single helicity geometry to prevent leading edges and reduce risk of macroscopic metal melting. Initial thermal simulations indicate the capability of this design to exhaust ~10 MW/m² of peak incident heat flux assuming a simple triangular distribution with 5 cm radial width and pulse length of 6 seconds. Select additional tiles, such as the neutral beam (NB) fast ion armor and known locations of significant NB shine-through, will also be armored with W panels. Armor in lower heat flux areas will likely consist of W coatings of thickness < 1mm on TZM or stainless steel. Coating technologies such as cold spray, atmospheric plasma spray, and chemical vapor deposition are being evaluated. Similar concepts are being considered for the new all-tungsten main wall of ITER. A joint operational assessment campaign is planned.

In later years, the divertor designs will include more innovative elements (e.g., negative triangularity) and advanced tungsten alloys. The DIII-D PMI research team, in conjunction with the U.S. fusion materials community, is examining a range of next-generation tungsten-based materials to enable this transition. Potential solutions include W-based multi principle element alloys, dispersoid-strengthened W, W fiber composites, and functionally graded W/SiC layers. Initial tests have been conducted using the DIII-D Divertor Materials Evaluation System (DiMES). Samples are exposed to H-mode plasma conditions with 30-40 Hz edge-localized modes (ELMs), allowing us to test the material response to transient heating. Some samples protrude above the floor, with the top surface tilted at a 10-15° angle towards the incoming plasma flux, facilitating a ~10× increase in power intercepted by the surface with no change to divertor plasma conditions. The level of damage to the angled samples, characterized by post-mortem microscopy and compositional analysis, varied widely between different materials tested; initial results will be presented.

Finally, progress on solutions to other key challenges for the transition of DIII-D to an all-metal wall will be addressed. These include, but are not limited to (a) changes to device start-up and conditioning procedures; (b) plasma control and manipulation techniques; (c) diagnostic upgrades and refurbishments. The DIII-D Wall Change-Out Project represents a significant leap forward for the reactor relevance of the facility and this two-stage plan ensures DIII-D's research outputs maintain their world-class status across this major transition.

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ADDRESSING CHALLENGES ASSOCIATED WITH MEASURING MIXED COMPOSITION FLOWS USING THERMAL-TYPE MASS FLOW METERS IN THE H3AT TRITIUM FUEL CYCLE

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The H3AT (Hydrogen-3 Advanced Technology) facility will consist of a closed-loop tritium fuel cycle pilot plant connected to a Tokamak Simulator and experimental laboratory, with a tritium inventory of up to 100g. The facility will have the capability to store, deliver, and purify tritium, and to provide the infrastructure and safety systems required to operate safely, as well as to dispose of the associated tritiated water waste.

To effectively simulate and control tokamak exhaust gases and other experimental gas profiles, Mass Flow Controllers (MFCs) play a vital role. The two main subsystems in which MFCs have a critical function in process control are the Tokamak Vacuum Simulator (TVS) and the Isotope Separation System (ISS).

Mass flow control is a well-established sector of process control and instrumentation with various designs to meet different physical, thermodynamic, and fluid dynamic properties. However, its application within a tritium fuel cycle is not straightforward due to challenges associated with varying process conditions such as temperature, pressure, composition, influence of tritium decay heat, and the requirement for materials to be tritium compatible.

The study presented here focussed on the application of MFCs within the TVS subsystem. The main operations of this system are the ITER-relevant Tritium Loop (TIRTL) operations, experimental glovebox operations, Neutral Injection Beam (NIB) simulation, and Disruption Mitigation System (DMS) simulation. All these operations require measurement and control of mixed composition streams to simulate specific discharge profiles. Initial studies suggest that thermal MFCs are the most suitable for these applications. They conduct their flow measurement via comparison of a temperature differential which is sensitive to variations in the specific heat capacity of the process gas relative to a calibration gas. The errors associated with measuring certain mixed gas flows within the TVS system were found to be significant.

The study aimed to quantify and experimentally validate the errors associated with using thermal-type mass flow controllers for flow measurement and control. The main method proposed to manage this issue is quantifying the errors associated with simulating various gas profiles and programming correction factors into the integrated control system. This will ensure the true mass flow is used for process control. Further studies will be conducted to investigate alternative solutions for MFC design to minimise errors.

First experimental characterization of the Magnetic Energy Storage and Transfer System (MEST)

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The Magnetic Energy Storage and Transfer system (MEST) is a technology designed to address the issues of high active power transients and huge reactive power demand in future large fusion reactors, such as DEMO. The MEST system employs an additional superconducting coil (sink coil) as an energy storage to sustain the operation of the tokamak coil (load coil). Through a switching section and a capacitor bank, it is able to transfer energy bidirectionally between the two superconducting coils, to provide the required current to the load coil.

The power peaks required by the load are flattened by the MEST system, therefore only the average active power has to be provided from the ac side via a conventional small rating power supply. Reactive power absorption is also reduced accordingly, and it can be further minimized by utilizing voltage source converters with an active front end.

A small-scale prototype of the MEST system is nearing completion, serving as a proof of concept and enabling the investigation of scaling the design to higher power ratings in view of DEMO. The prototype operates with resistive sink and load coils and follows a modular approach for the switching section (combined with the capacitor bank), with two modules in total. The modules are designed for the operation at 2 kV and 2 kA and can function independently or be connected together to achieve higher ratings. This prototype will permit the evaluation of the advantages and limitations of the specific topology, examine various configurations, test different control strategies, and help identifying any unexpected experimental findings.

The first tests focus on verifying the prototype's functionality, particularly the integration of the components, and assessing the system's ability to effectively control the currents in both coils. This work presents the results of the initial tests, the first characterization of the circuit and the validation of the developed numerical models with experimental data.

Design and commissioning of control algorithms for optimal ECRH power injection in TCV's plasma

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The Tokamak à Configuration Variable (TCV) features an Electron Cyclotron Resonance Heating (ECRH) system equipped with (currently) five microwave transmission lines. The three lateral ones carry microwaves at the 2nd harmonic (X2) of the electron cyclotron frequency. This harmonic is accessible from a tokamak's low magnetic field side, simplifying power delivery and thus plasma heating. One pair of polarizing mirrors is included in each transmission line. These grooved mirrors, whose groove's depth depends on the microwave wavelength, can be rotated about their normal axis to change the beam polarization. From a mechanical point of view, they can be modelled as single-degree-of-freedom (SDOF) systems affected by harsh disturbances, among which friction plays a major role. Controlling these devices properly is crucial to maximize the injected power's absorption and prevent stray radiation from damaging in-vessel components. This work regards the design of local controllers for real-time control of the four polarizers in two of the X2 transmission lines of TCV. Before this work, the polarizing mirrors' positions were set manually between pulses, with no real-time tracking of the optimum polarization angles, which depend on plasma parameters and the launching angle. A PID controller is designed and commissioned to overcome this limitation. Real-time polarization control later allowed the possibility to study the effects of microwave polarization on plasma; dedicated experiments were run to assess how polarization influenced both plasma temperature and stray radiation, with the final goal of validating the current theoretical model used to compute the optimum values for microwave polarization angles. Further experiments will aim at deriving an experimental model relating temperature and polarization and stray radiation and polarization, to be used in the upper levels of the control architecture.

Designing irradiation-tolerant TaTiVW refractory high-entropy alloys with defects simulation and heavy ion irradiation

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Body-centered cubic (BCC) single-phase solid-solution Refractory high-entropy alloys (RHEAs), comprising Ta, Ti, V, and W, are proposed as candidate structural materials for nuclear fusion, subject to specific downselecting constraints. The primary focus of this research lies in investigating the defect recombination capability under the hypothesis that variations in the potential energy landscape (PEL) determine the irradiation tolerance of RHEAs. Molecular Dynamics (MD) will optimize the composition of RHEAs to achieve the most PEL heterogeneity. Heavy self-ion irradiation will then be employed to compare various samples and validate the simulation findings. To ensure consistent defect mobility, the irradiation temperature will be maintained at one-third of each sample's homologous temperature (Tm). Additionally, In situ ion irradiation transient grating spectroscopy (I3TGS) and Transmission Electron Microscopy (TEM) will be employed to evaluate thermal elastic properties and microstructures to assess radiation-induced damage.

Characterization of Astral System's Mark I Fusion Reactor

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Astral Systems is pioneering the development of fusion reactors for a wide range of applications, including the production of medical radioisotopes, boron neutron capture therapy, neutron activation analysis, nuclear materials analysis, nuclear security, and fusion-driven subcritical fission power concepts. The innovative fusion reactor, Mark I, is equipped with two electrodes connected to opposing polarity potentials, powered by an 18-kW direct current supply, and managed through a LABVIEW-based graphical user interface. The reactor operates by confining gases like Deuterium (D) or Tritium (T) within a vacuum vessel under carefully controlled voltage and current conditions. The vessel, filled with gas at pressures of a few pascals, undergoes ionization through applied voltage and current, initiating plasma formation and nuclear fusion. This process generates neutrons with energies of 2.45 MeV for DD fusion and/or 14.1 MeV for DT fusion.

To ensure safety and efficiency, simulations were conducted to design a radiation bunker capable of controlling the radiation dose around the reactor. The system is housed within a shielding bunker featuring a 50cm thick water barrier, sufficient to shield up to 1e8 n/s DD neutrons. These measures ensure minimal radiation exposure while enabling the reactor to operate at an optimal capacity of 3 mSv/h. Mark I's unique design enables significantly enhanced fusion rates using a novel lattice confinement architecture. With dimensions of 50 cm in height and 20 cm in diameter, the reactor has been successfully designed, fabricated, commissioned, and tested. It represents a breakthrough as a multi-state fusion device. The reactor's parameters have been extensively characterized, with relationships such as the Paschen curve, gas pressure versus applied voltage, system performance at differing settings of the water-cooling temperature, and current-voltage dynamics mapped. Additionally, key outputs like the voltage-neutron yield and current-neutron yield curves have been determined, alongside many other variables which have been tested.

The performance of the Astral Mark I reactor, utilizing lattice confinement fusion, the radiation bunker simulation results, combined with detailed reactor characterization data, underscore the Mark I fusion reactor's potential to revolutionize multidisciplinary applications. This breakthrough demonstrates a promising pathway toward practical fusion technologies and their integration into critical fields.

Fault Analysis and Protection Logic for Bridge Arm Thyristors in ITER PF Converters

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Thyristor failures are the main factors for converter faults. For an anti-parallel system with 12 parallel thyristors in one single bridge arm, faults could be worse, particularly for ITER poloidal field (PF) converters. The bridge currents could be out of control, threatening the safety of related equipment. This paper investigates the mechanisms behind four types of thyristor faults firstly. Based on the reference voltage and bridge current balance control of PF converter modules, the potential evolution of certain faults is further discussed. Since some faults might remain undetected as they do not cause immediate overcurrent, secondly, existing protection logics for the PF converter are presented and analyzed based on observed fault phenomena and underlying mechanisms. Thirdly, to ensure timely detection and clearance of these faults, the protection logic is optimized using existing sensor configurations. Finally, simulations are employed to verify the fault mechanisms, and the protections are validated.

A Physics-Informed Neural Network for Fast Equilibrium Calculations in Tokamaks

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Tokamak operation relies on accurate and efficient free-boundary equilibrium (FBE) solvers. Traditional solution methods, such as those based on finite-difference and Picard iteration [1], can be computationally expensive. This work introduces a novel approach using physics-informed neural networks (PINNs) to solve both direct and inverse FBE problems. The direct problem determines the 2D poloidal-flux map given plasma conditions and poloidal-field (PF) coil currents. The inverse problem calculates the required PF coil currents and the 2D poloidal-flux map for specified plasma conditions and target shapes. The PINN surrogate models, first developed to address the direct problem [2] and now to address the inverse problem, employ fully connected multi-layer perceptron (MLP) architectures and incorporate the Grad-Shafranov equation as a physical constraint in their loss functions, ensuring consistency between plasma conditions, PF coil currents, and the resulting 2D poloidal-flux map. Extensive training datasets were generated to encompass a broad range of plasma configurations, including variations in plasma current, poloidal beta, major and minor radii, triangularity, and elongation. The performance of the inverse-mode PINN developed in this work was benchmarked against conventional FBE solvers, demonstrating high accuracy in replicating the outputs of the FBE solver and achieving a low root-mean-squared error (RMSE) for the magnetic flux maps. Furthermore, the direct and inverse-mode PINN models were integrated into a unified solver capable of handling both types of equilibrium problems. This unified framework significantly enhances computational efficiency, completing calculations in a fraction of a second compared to tens of seconds for traditional iterative solvers. By combining speed and accuracy, the integrated PINN-based solver offers a robust tool for real-time control and optimization applications in tokamak research.

[1] Song, X. et al. (2024). Plasma, 7(4), 842-857.
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Power supply control using low-cost Raspberry Pi Pico Microcontroller for a tokamak

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A modular high current switching power supply system is currently under development for the Columbia University Tokamak for Education (CUTE) experiment in construction at Columbia University. Each of the 28 independent poloidal/OH and toroidal field (2 parallel circuits) coils will be driven by 56 0.9 MVA (1kA, 900V) insulated-gate bipolar transistor (IGBT) H-Bridge switching power amplifiers (SPAs) previously used for the HIT-SI family of devices. A new digital control system utilizing the Raspberry Pi Pico microcontrollers to control each high-power SPA allows precise driving of currents for each poloidal and toroidal field coil. Feed-forward operation is planned for initial operations with proportional-integral differential (PID) feedback for coil current using onboard digitizers and a minimum control loop period of 20µs. Future updates to the system will support realtime control through serial datalinks from a central control system. Initial low power tests have demonstrated the Raspberry Pi Pico and PID control program's ability to consistently output desired current waveforms with an inductive test load. Design, implementation, and plans for the overall power system will also be presented.

Developing an Ultrasonic Measurement System for Flowing Liquid Lithium Plasma Facing Components

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The performance of a flowing liquid lithium divertor in fusion environment depends on its ability to effectively withstand the heat from the plasma. The greatest advantage of flowing liquid plasma-facing components (PFC) is the absence of a hard limit to the heat flux it can take without failing. The practical limit is primarily defined by the flow speed of the liquid surface. This value often directly depends on the average flow rate of the liquid metal in the manifolds circulating liquid metal in and out of the fusion reactor vessel. Existing techniques to measure the average manifold flow rate struggle at monitoring the open surface flow of liquid metal PFCs, due to the highly corrosive nature of lithium as well as other engineering restrictions. At University of Illinois Urbana-Champaign, we explore the utilization of ultrasonic techniques to measure the velocity distribution in the flow cross-section. In this work, we discuss a number of engineering challenges associated with building a liquid lithium ultrasonic velocity measuring sensor, such as the high acoustic mismatch between the liquid lithium and fusion-relevant structural materials. We present the data produced by the multiphysics simulations and early experimental results with water. And an initial concept design for a measurement device is presented.
Tokamak Energy Ltd. sponsors the research at UIUC to construct a liquid lithium loop named APOLLO to develop liquid lithium PFCs, particularly a divertor plate. Developing the abovementioned technique is necessary for enhancing APOLLO's experimental capabilities.

Design and Prototype Testing of the Columbia Stellarator eXperiment (CSX) Non-Planar HTS Magnets

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The increase in the widespread use of high-temperature superconductors (HTS) has contributed to a surge in fusion companies aiming to achieve sustainable burning plasmas through magnetic confinement. This technology allows for stronger magnetic fields in more compact devices, with wider operational margins than with conventional low-temperature superconductors. However, the HTS tape form factor makes it better suited to planar magnets, and the adaptation to non-planar stellarator magnets thus presents several challenges. We present work being done to design and test non-planar HTS magnets for the Columbia Stellarator eXperiment (CSX), a quasi-axisymmetric device composed of two optimized non-planar interlocked HTS magnets and two copper poloidal field magnets. At present, prototype non-planar HTS magnets have been successfully tested in liquid nitrogen, with ongoing efforts to achieve higher on-axis fields of 0.5 T in a cryogenic test stand. Concurrently, the full-scale interlocked magnets are being designed and optimized. Prototype and eventual full-scale magnets consist of HTS tape wound in channels on 3D-printed aluminum bobbins and vacuum solder-potted. A gimballed winding mechanism maintains constant tension and mitigates strain during the winding process. The prototypes are conductively cooled to 20 K target temperatures using a coldhead in a bell jar test-stand, with additional heat shielding to optimize cooling. Diagnostics and external controls are interfaced with LabView and include a high-current power supply, scanning Hall probes, temperature sensors, and ion pressure gauges. This work aims to enable the construction of a strain- and field-optimized university-scale HTS stellarator, addressing critical engineering challenges in the adaptation of HTS technology for stellarator configurations.

Cooling channel optimal design and flow uniformity analysis of divertor

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As a vital heat-load component in Tokamak, it is necessary to cool the divertor sufficiently and efficiently. At present, a divertor module is composed of several plasma facing unit (PFU), resulting in a large pressure loss. At the same time, due to the structural differences between different PFU, the cooling water flow distribution in each PFU will be different, and in serious cases, the local heat transfer capacity will be reduced. In this paper, a variety of divertor cooling channel structures are proposed, and the flow analysis of each structure is carried out. The pressure drop and flow distribution in PFU of different structures are compared, and the structural design is optimized. The results show that after structural optimization, the flow pressure drop of a divertor module can be reduced from 1.41MPa to 1.24MPa, and the maximum deviation of flow between each PFU is reduced to less than 10%. The maximum deviation of flow in the vertical and horizontal sections of the outer target are 6.5% and 8.8%, respectively. The maximum deviation of flow in vertical and horizontal sections of the inner target are 6.8% and 0.6%, respectively. The results of this study will be helpful for the future design of divertor and provide guidance for its structural optimization.

High-Prandtl MHD Turbulence Modeling for Fusion Liquid Blankets

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Molten salts (MS) have emerged as promising multifunctional materials for the liquid breeding blanket in nuclear fusion reactors due to their radiation resistance and heat transfer capabilities. In fact, these materials can simultaneously perform the duties of breeder, neutron multiplier, and coolant. However, the flow of MS is affected by magnetohydrodynamic (MHD) effects in the presence of magnetic fields, altering flow patterns and posing challenges in the design of the reactor cooling system. Most commercial Computational Fluid Dynamics (CFD) software typically use standard turbulence closures that do not model the fluctuations of the Lorentz force in the turbulent variables model equations. In addition, the high Prandtl number (Pr) of MS adds complexity to heat transfer modeling.
This study addresses these limitations by evaluating the applicabiltity of MHD turbulence models specifically tailored for MS applications in fusion reactors. The implemented models incorporate the effects of MHD interactions and are validated against several benchmark cases and experimental datasets representative of fusion reactor conditions. The results highlight the importance of accurate turbulence modeling to better represent fluid-magnetic interactions in MS blankets, which are critical for improving heat transfer, neutron shielding, safety, and tritium breeding. Conducted within the OpenFOAM framework, this work contributes to open-source computational tools to support the design and optimization of innovative nuclear fusion energy systems.

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Implementation of a high resolution two-thousand signal acquisition and preprocessing system for the magnetic and electrical measurements of RFX-mod2 experiment.

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Most of magnetic fusion devices rely mainly on coil loops as the primary type of magnetic sensors, offering precision, reliability, and robustness. However, this type of sensors provide a derivative signal which need to be time-integrated. Traditionally, analog integrators have been employed due to their wide dynamic range, but they present a number of complexity challenges. For example, a separate channel for the derivative (dB/dt) signals is also required for proper measurement and control of fast events, plasma instabilities, and magnetic turbulences. Another crucial issue is related to the hostile EMI environment, subject to constant disturbances and potentially damaging overvoltage spikes, in which the system is operated.
In this work, we present the architecture and the subsequent validation tests of a unified system developed for RFX-mod2, designed to manage 1736 magnetic (pick up coils, saddle coils and Vloops) and 480 electrical signals (current and voltages of magnetic field coils), for a total of 2216.
Each channel is based on high-resolution analog-to-digital converters (1 MS/s, 20-bit ADCs) that eliminates the need for analog integrators and the auxiliary acquisition channel, simplifying the overall system and offering electrical characteristics comparable to good analog integrators. Moreover, to ensure an adequate noise immunity and rejection, each acquisition channel is galvanically isolated, effectively eliminating the ground loops disturbances generated by the experiment's magnetic fields.
The system architecture has been implemented to be easily scalable: the basic element consist of 6U boards, each one being an autonomous system housing 12 input channels and its own SOC-FPGA, then combined in a 144 channel 6U sub-rack. Each board simultaneously provides three essential functionalities needed in a fusion facility: a timing synchronization decoder, transient recording of full-speed ADC data, and continuous Ethernet UDP/TCP transmission of pre-processed and subsampled signals for real-time control system. This comprehensive approach allows for efficient data acquisition, analysis, and integration into the experiment's control framework.

High Field Non-Nuclear Blanket Component Testing User Facility to Complement International Capabilities

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Preconceptual design for a non-nuclear U.S. high field blanket component testing facility (BCTF) will be presented, considering present and planned international capabilities and needs. A critical and low technology readiness level (TRL) component to tritium and power-producing fusion concepts, maturation of the blanket from the current low TRL (~2-3) to high TRL (~6-7) is required to ensure performance requirements will be met, and before approval by safety and regulatory stakeholders. BCTF requirements have been developed with community input and consist of (i) provide magnetic field and gradient prototypic of the high field side of toroidal devices with field up to 10 T, (ii) provide breeder loops at relevant temperatures (300-700 C), (iii) provide auxiliary coolant and heat rejection loops (e.g. high pressure water, 8 MPa helium) and (iv) provide surface (~MW/m2) and simulated volumetric heating (~MW/m3). These requirements produce the relevant non-nuclear environment for qualifying meter-scale blanket components at the appropriate electromagnetic, thermal, and mechanical loading, as well as high MHD Hartmann, Reynolds and interaction parameter values for liquid metal breeding blanket concepts (Li, PbLi). Dual-cooled PbLi/He and Li/He are complementary to international efforts. Hydrogen production simulants are presently under consideration. BCTF design incorporates significant onsite materials, infrastructure and capability developed during fabrication of the ITER central solenoid (CS). Excess Nb3Sn CS conductor in spools and wrapped as a six-layer hexapancake, fully integrated 1kW@4K supercritical helium cryoplant, 24kL/s vacuum, and 50kA@15V power supply are present in the facility and integrated into the BCTF design.
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Optimization of Fast-Response Power Supply System Based on PWM Control

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Fast-response power supply systems play a vital role in modern industry, and their dynamic response speed and output accuracy directly affect the performance of equipment. Aiming at the problems of slow response speed and low accuracy of traditional fast-response power supply systems, this paper proposes an optimization scheme based on PWM control. The scheme first analyzes the basic principle of PWM control and its application advantages in fast-response power supply systems, and then improves the fast-response power supply system from two aspects of hardware circuit design and control algorithm optimization. In terms of hardware circuit design, high-frequency switching devices and low-loss magnetic components are adopted to improve the switching frequency and efficiency of the system; in terms of control algorithm optimization, advanced control strategies such as fuzzy control and adaptive control are introduced to improve the dynamic response speed and anti-interference ability of the system. Finally, the effectiveness of the proposed scheme is verified by simulation and experiments. The results show that the optimized fast-response power supply system has faster dynamic response speed, higher output accuracy, and stronger robustness, and can better meet the application requirements of modern industry.

Improving the Accuracy and Uncertainty Evaluation of Nuclear Fusion Energy Prediction Using Bayesian Neural Network(BNN) and Ensemble Learning

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This research reports on predicting fusion output using Bayesian Neural Networks (BNN) enhanced by ensemble learning. Accurate prediction of fusion output is essential for optimizing the performance, safety, and control of future fusion reactors. While traditional machine learning methods have shown promising results in this field, they often lack the ability to quantify uncertainty associated with predictions. BNNs provide a principled framework for incorporating and propagating uncertainty, enabling more robust and reliable predictions, particularly in extrapolation to regions where training data is limited.
Our approach analyzed experimental databases used in confinement scaling, employing an ensemble of BNN architectures to predict fusion output based on various plasma parameters. We investigated different inference methods within the Bayesian framework and evaluated their impact on prediction accuracy and uncertainty quantification. The model's performance is assessed using standard metrics such as Mean Squared Error (MSE), Root Mean Square Error (RMSE), Mean Absolute Error (MAE), and the coefficient of determination (R2). Our results show an R2 of 0.64. However, when extrapolating towards target design values for ITER and demonstration reactors, the uncertainty in predictions increased significantly, highlighting the need for further model improvements and additional training data to more accurately evaluate the probability of achieving these targets.
The findings demonstrate that BNN enhanced with ensemble learning can provide not only accurate point predictions of fusion output but also valuable estimates of the probability of achieving target goals. To quantify this uncertainty, we determinate the 2-standard-deviation (2std) band width the predicted values. This representation provides a general indicator of prediction variability, which is critical for quantifying the probability of achieving ITER and demonstration reactor targets . It further aids in informing risk assessments and decision-making in fusion device operations.
Finally, we explored the sensitivity of BNN predictions to different input features, gaining insights into the key plasma parameters influencing fusion performance. This study advances research into the application of advanced machine learning techniques to address critical challenges in fusion energy development. In particular, the quantified uncertainty represented by the 2std band offers significant advantages over traditional methods, clearly quantifying the probability of achieving ITER DEMO and demonstration reactor targets. Future work will focus on improving model accuracy and identifying the necessary experimental devices and plasma data to reduce uncertainty.

Development of the Structural Simulator for the calibration of the Virtual Reality environment of the ITER Remote Handling Control System

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Remote maintenance represents an important function with many challenges in the design and operation of a fusion reactor. The ITER project will demonstrate the design, integration and operation of a Remote Handling System (RHS) able to perform both pre-planned tasks and recovery/rescue operations in case of failure or abnormal conditions. The ITER Remote Handling Control System (RHCS) is based on a man-in-the-loop concept due to the complexity of the tasks, which are generally not automated, while executed step-by-step by the operator who has to make decisions and may need to take direct control of the equipment. In order to make more informed and aware decisions while supervising or teleoperating RH equipment movements, the operators are provided with both real images coming from on-site cameras and virtual renderings, thanks to the Virtual Reality (VR) System. The VR is part of the High Level Control System (HLCS) at the operator workcell in a distant location, and is monitoring Remote Handling devices through a real-time network connection. Since the VR System should be as accurate as possible in representing the real robot configuration to effectively help the teleoperators and inform them in case of possible collisions, a Structural Simulator is required as part of its features to model and visualize the dynamic effects causing deviations of the real robot configuration from the nominal one, guaranteeing the calibration of the VR environment to match the real world. The present paper introduces the development activities of the Structural Simulator for the Virtual Reality System of the ITER RHCS. In particular, the strategies to calibrate the VR environment by representing in the robot model the kinematic and non-kinematic errors – i.e., the geometric errors caused by the clearances present in the joints due to manufacturing tolerances and the non-geometric errors due to the joints/links flexibility, respectively – are presented. The calibration strategies are applied to the case of the ITER Cassette Multifunctional Mover (CMM) prototype in DTP2 facility at VTT Technical Research Centre of Finland. After the import of the CAD models in VR environment and the implementation of the kinematic and dynamic data, the kinematic model is corrected by adding error parameters, transformation matrices at joints, or even virtual joints, to display the real robot configuration considering the kinematic and non-kinematic errors. The location, type, and entity of the errors to be corrected are either measured through test sessions at DTP2 facility or predicted through analyses both of joints/links flexibility through FEM and of joints internal manufacturing tolerances documented in technical drawings. Tests are then carried out to verify if the corrected VR model matches the real one complying with the target accuracy, basing on the data measured on the CMM physical prototype at DTP2 facility.

Fast and high-fidelity computation of neoclassical toroidal viscosity torque by leveraging machine learning surrogate for drift kinetic equation solver

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Neoclassical toroidal viscosity (NTV) torque is a crucial source for toroidal momentum transport in tokamaks and has significant influence on plasma instability and performance. Drift kinetic equation (DKE) is the basic equation governing the physics law of NTV, and also the time efficiency bottleneck of NTV numerical calculation. In this paper, a machine learning surrogate for solving DKE (DKE-ML) is developed and coupled with phase space integral calculation to realize fast and high-fidelity NTV torque computation. The main development process consists of four steps: First, to obtain a comprehensive dataset for DKE, extensive numerical calculations of NTVTOK and MARS-F codes were performed under various plasma conditions of Experimental Advanced Superconducting Tokamak (EAST). Second, the surrogate model of DKE is designed and developed based on machine learning method. Third, a novel and efficient hyper-parameter optimization algorithm is used to optimize the model performance. Finally, the surrogate model DKE-ML, taking 1/5 computational time of the original finite difference based DKE solver, is coupled with phase space integral process to realize NTV torque computation. The coefficient of determination of DKE-ML is higher than 0.95, demonstrating the high prediction accuracy. By coupling DKE-ML with phase space integral calculation, the coefficient of determination of NTV torque prediction is higher than 0.99.

Advancements in the Continuous Pellet Injection of Hydrogen Isotopic Mixtures

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Pellet injection is the principal technology developed for continuous efficient fueling to enable and control a burning fusion plasma. The injection of solid pellets of hydrogenic isotopes (H2 and D2) into plasmas been widely deployed on fusion experiments and has extensive development history. There are key aspects of this technology that remain to be developed and demonstrated, particularly the use of DT for continuous pellet fueling. TPI –1 at the PELIN Laboratory has demonstrated continuous extrusion of DT mixtures, but only in low mixture concentrations, and low flow rates. It has not been established how to operate a continuous screw extruder with high concentration mixtures and high flow rates.

A primary challenge in extruding isotopic mixtures of hydrogen is due to the highly temperature dependent dynamic shear strength of the isotopes. The shear strength of hydrogen isotopes is known to increase strongly as the temperature decreases below the triple point. An extruder must operate below the triple point temperature of the lightest isotope in the mixture to maintain a solid, but this temperature results in a high shear strength for the heavier isotope. Depending on the mixture ratio, this will result in much higher torque and thrust forces produced within the extruder than are currently exhibited in single isotope extrusions operating at a specific temperature optimized for that isotope. These enhanced loads on the extruder can cause leaks at the screw shaft seal, nozzle seal, and shorten the lifespan of the extruder drive components.

This paper will detail the recent advancements at the Oak Ridge National Laboratory of using HD as a DT surrogate mixture in a single-screw extruder-based pellet injector. Insights into injector operation best practices and experimental results will be presented.

A High-Power Pulse Switch Module with High Repetitive Capability for Compact Torus Injector

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Compact Torus Injector (CTI) is one of the most promising central fuelling methods for the future fusion reactor, which must be implemented in a repetitive mode. The realization of CTI repetitive operation mainly depends on its power supply system, and the pulse switch is the technical bottleneck which restricts its further development. In response to this urgent need, two sets of high-power pulse switch components have been developed. One set, based on the ignitron technology, and a peak current of 330 kA, a half-wave pulse width under 20 µs, a pulse frequency of 2 Hz, and a total of 20 pulses operation have been achieved in experiments. However, the drawbacks of this kind of switch are quite obvious. Specifically, its coulomb number is limited, and it cannot operate stably for a long time.

The other set of switches is developed based on high - power semiconductor switches. It is composed of two series - connected semiconductor switches: a GTO-like (Gate Turn-Off) thyristor and a Fast Recovery Diode (FRD). The switch module adopts a low - inductance coaxial structure. The GTO-like thyristors are employed to control the turn - on of the main circuit, and the FRDs are used to protect the GTO-like thyristors by turning them off in advance. A pulse power discharge test platform for the semiconductor hybrid switch module is designed. When the applied voltage is 10 kV, the peak current of the discharge circuit reaches 200 kA, the half-wave width of the current waveform is 15 µs, the repetitive rate is 10 Hz (under a peak current of 100 kA), and a maximum of 100 pulses have been achieved. This indicates that the FRDs can effectively prevent the switch module from being damaged by reverse overvoltage, and the switch module can enhance the parameters and reliability of the power supply. In the future, the operating conditions with large currents can be met by increasing the number of parallel connections.

Erosion and H Retention Measurements of W and SiC Materials for RF Antenna Applications

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Antennas in fusion reactors generate radio frequency (RF) fields and plasma sheaths that interact with the local plasma which can result in increased erosion of the antenna structure relative to the typical plasma sheath formed in front of plasma-facing components held at a constant electrical potential. The RF Plasma Interaction Experiment (RF PIE) at Oak Ridge National Lab (ORNL) is used to simulate this RF antenna structure. RF PIE is an Electron Cyclotron Resonance microwave-based plasma source (2.45 GHz, <5 kW) with an RF biased electrode. RF PIE can provide relatively clean spectral plasmas and a simplified viewing geometry with a large solid angle making spectral measurements easier to capture. One side effect of RF fields and the RF sheaths they produce is increased sputtering rates. Antennas are commonly made of tungsten which can withstand the high temperatures from a fusion plasma, but as a high-Z material it will result in excessive radiative power losses in the plasma when introduced as an impurity. Alternatively, silicon carbide (SiC) is a lower-Z material that will result in lower radiative emission and also has been demonstrated to be less susceptible to neutron damage. This presentation shows a comparison between pure W, pure SiC, and a W-SiC composite material with regards to their manufacturability and suitability for an RF plasma environment. The composite material is being manufactured using spark plasma sintering, a hot-pressing technique involving rapidly heating a powdered sample mixture while it's under pressure. This technique allows for different composite materials to be created with varying atomic concentrations of W and SiC. After each material is annealed, it is exposed to a He or D plasma (10^24 m-2 fluence and 10^18 m-2s-1 flux density) and biased with a RF voltage ranging 0-500 V. Sample exposures are performed at both room temperature (22 C) and at 600 C.
Two important factors to consider when choosing an antenna material are the material's ability to resist erosion and resist retaining large amounts of hydrogen isotopes. Micro-trenches (10 x 10 x 6 um) are etched into the samples along with fiducial depth markers along one wall. Erosion is calculated using SEM images of the trenches before and after exposures to measure the depth change of the trenches using the fiducial markings. A spectrometer is used periodically during the exposures to measure sputtering rates and confirm results from the trenches. Thermal desorption spectroscopy (TDS) is used after the exposures to measure the hydrogenic retention of each sample. Initial results from pure tungsten TDS measurements followed the expected pattern of an increase in H retention with an increase in the RF bias voltage applied to each sample. As a material incurs more damage from the plasma, there are more places for hydrogen to be trapped in the surface. Further comparisons with pure SiC and the W-SiC composite material will be discussed.

Effect of Y_2O_3 nano-particles on microstructure and properties of reduced activation ferrite / martensite steel fabricated by selective laser melting

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In recent years, human beings are constantly seeking solutions to solve the energy crisis problem, and nuclear fusion energy has become one of the most promising clean energy sources. The generation of nuclear fusion energy needs to be established on the construction of fusion reactors, and the urgent problem to be solved in order to achieve fusion power generation is the materials issue, especially for plasma-faced tungsten and reduced activation ferrite/martensite steel (RAFMs). In this paper, the first wall structural material of blanket (RAFMs) was investigated and a new fabrication method was proposed. The RAFMs powder doped with 0.5wt.% Y2O3 particles (oxides dispersion strengthening steel, ODS steel) were manufactured by selective laser melting (SLM) and the samples were regulated by normalizing and tempering heat treatment. The effect of Y2O3 particles on the microstructure and properties of RAFMs was analyzed. It was observed by X-ray diffraction that the phase of ODS steel was mainly Fe-Cr phase, oxides with low content was unable to be detected, the diffraction peak was difficult to separate from the matrix so it was basically consistent with RAFM steel phase. In the microstructure of ODS steel, a small number of micro-cracks occurred and were concentrated around the oxides, which was mainly related to the high melting point of Y2O3. The interface between Y2O3 and the matrix was prone to stress concentration due to the difference in properties during the forming process, and the direction of crack expansion was perpendicular to the direction of the lath boundary. The TEM micromorphology showed that Y2O3 particles and Yi-Ta-O oxides are dispersed in the matrix. These oxides could effectively nail the dislocations and played an important role in improving the mechanical properties of RAFMs. In this paper, the changes of grain before and after iron ion irradiation were also compared. The grain increased with the irradiation dose both of RAFMs and ODS steel, while the increased degree of ODS steel was smaller than that of RAFMs which was proved that Y2O3 particles can nail grain boundaries and hinder grain growth, thus improving the radiation resistance. The results of stress-strain curves at room temperature showed that the average tensile strength of ODS steel reached 983 MPa and about 774 MPa after heat treatment. The average tensile strength of RAFMs was about 911 MPa and 665 MPa after heat treatment, which proved the dispersion strengthening effect of Y2O3.

Development of the hybrid joint for the connection between superconducting conduit conductor

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The development of reliable joining technology for superconducting conductors like as cable in conduit conductor is critical for advancing fusion energy systems. In this study, a method to join superconducting conductors of uniform dimensions while ensuring operational stability. The joining process involves partially removing the superconducting strands on both sides and embedding the remaining strands onto a copper core, followed by soldering. This approach significantly reduces contact resistance. Furthermore, to ensure cryogenic stability, high-temperature superconducting (HTS) strands are stacked within the copper core, allowing current to flow through the HTS strands in the event of coolant temperature rise. This joining technology not only provides robust electrical and mechanical connectivity but also addresses key challenges in maintaining superconducting performance in harsh fusion environments. In this paper, we will present manufacturing process and the results of the contact resistance measurements.

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Generative AI-powered assistant for pattern recognition in TJ-II/W7X stellarators

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After an exhaustive study of generative AI services and the selection of the appropriate providers and technology, the current proposal to use a secure generative Artificial Intelligence (AI) platform with nuclear fusion-specific data is currently underway for TJ-II and W7X stellarators. This includes data migration, integration, and the study of the expected feedback using chatbots. The creation of the hybrid cloud and its connectivity, as well as data integration and the use of a vector database, is already finished. Subsequently, and in parallel with the processed data from TJ-II and W7X, specialized large language models (LLM) were programmed, trained, and validated to compare them with other self-learning models generated by complex algorithms and processes commonly used in both TJ-II and W7X devices. From here, the ultimate goal is to develop relevant use cases related to pattern recognition and image classification issues that will be addressed using generative AI services and, if necessary, reinforcement learning of the previously generated LLM models. Additionally, this task pursues to use the conversational interface as a common framework for extracting significant and hidden insights that can be scientifically validated

Electronic detection for MRSnext at the Pacific Fusion experimental facility

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Monday Posters 2, Lobdell (Building W20 Room 208), June 23, 2025, 2:00 PM - 3:30 PM

The Magnetic Recoil Spectrometer (MRS) has been used at the OMEGA Laser and National Ignition Facilities for over a decade to measure the neutron spectrum in inertial confinement fusion (ICF) experiments. This allows important parameters to be diagnosed, such as areal density, yield, ion temperature, and alpha heating, which are essential for evaluating implosion performance. While the current MRS, with a CR-39 array as the detector backend, has been incredibly useful in support of the ignition experiments at the NIF, it is not ideal for higher-repetition-rate scenarios. Moreover, the etching and scanning procedure used to process the experimental data from the CR-39 typically takes weeks post-experiment. To substantially improve signal-to-background and data turnaround time, the next-generation MRS (MRSnext) being designed for the experimental facility at Pacific Fusion will use electronic detection. This poster discusses design options for the electronic backend readout of MRSnext, which include silicon strip detectors, segmented scintillators coupled to photomultiplier tubes, Si-PIN photodiodes, and a CVD diamond array coupled to an oscilloscope. Using synthetic MRSnext signal to model the performance of each design, a study was performed to determine which design best suits the experimental needs of Pacific Fusion.

Design and test of a waveguide filter for reflected electromagnetic waves

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Monday Posters 2, Lobdell (Building W20 Room 208), June 23, 2025, 2:00 PM - 3:30 PM

A high power gyrotron system is used for plasma production and sustainment by electron cyclotron resonance heating (ECRH) in fusion reactors. An ECRH system in the Large Helical Device utilizes gyrotrons with megawatt output power. This megawatt electromagnetic wave transmitted through a circular corrugated waveguide is partly reflected back to the gyrotron and can be a cause of unstable gyrotron oscillations. The reflected beam may also propagate at an angle relative to the waveguide axis. In order to largely reduce the reflected beam intensity while retaining as much of the forward-propagating gyrotron beam as possible, we aim to optimize the length of a filter consisting of a gap with absorbing walls between waveguide sections in a transmission line. A semi-analytical model is developed using diffraction theory to calculate the transmitted power of the HE11 mode as a function of filter gap distance for a wide range of beam angles, including a perfectly aligned beam to approximate the forward gyrotron beam. Before implementing this filter, we use a vector network analyzer to measure the scattering parameters for an experimental mock-up of this filter with varying gap distances. These sophisticated models are used to inform the gap filter design and length optimization.

Harnessing Uncertainty Quantification for Automated Anomaly Detection in Fusion Diagnostics

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Monday Posters 2, Lobdell (Building W20 Room 208), June 23, 2025, 2:00 PM - 3:30 PM

Fusion experiments produce vast amounts of diagnostic data collected from multiple shots across heterogeneous diagnostics, covering diverse spatial and temporal scales. As fusion research advances, the ability to detect anomalous data automatically is crucial for optimising performance and ensuring data reliability. We propose a framework that leverages underlying trends within this extensive dataset and uses uncertainty quantification to achieve robust and automated anomaly detection.

Data processing routines that convert raw measurements into usable quantities are susceptible to interference, fatigue, calibration drift, and spurious events, which can introduce anomalous data points. These anomalies are often buried within large datasets, and require manual, time-consuming extraction efforts. Without intervention, multi-shot data analysis workflows risk misidentifying useful data and discarding entire plasma shots due to a few anomalous data points. This limits the use of historical data and can obscure critical trends. Moreover, undetected anomalies may mask underlying issues, increasing the likelihood of repeated failures in future experiments. This challenge becomes even more critical for future reactors, where proposed digital twins rely on good-quality data.

To address these challenges, we propose an uncertainty quantification framework based on Gaussian Processes (GPs), offering a robust, data-driven approach to anomaly detection. GPs capture complex data characteristics, such as length scales and periodicity, while naturally accommodating aleatoric uncertainty (inherent noise) and epistemic uncertainty (irregular and missing data). Their probabilistic nature, inherently providing uncertainty quantification, allows credible intervals to be established, enabling the identification of anomalous data points beyond expected system fluctuations.

Accurate anomaly detection using GPs benefits from dense data coverage to distinguish genuine anomalies from system noise and reproducible physics phenomena. Correlations across spatial and temporal dimensions, including multiple shots, contribute to this data coverage. Despite measuring different quantities, substantial correlations exist between distinct diagnostics. This allows anomaly detection in one diagnostic to leverage complementary data from others. However, the volume of the resulting data makes standard GP methods computationally infeasible.

To overcome this limitation, we propose a hybrid workflow combining GPs with deep learning techniques to process large, heterogeneous diagnostic datasets efficiently. By learning latent representations of the data, our framework enables self-prediction of each diagnostic signal, with uncertainty thresholds facilitating automatic anomaly detection. This approach not only streamlines the cleaning of historical data but also enables deeper investigation into the root causes of anomalies, ultimately enabling more reliable fusion operations and maximising experimental output.

Characterization of full-domain first wall heat loads for SPARC X-Point Target experiments using SOLEDGE3X-EIRENE

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Addressing plasma exhaust is a critical challenge for the ARC power plant and other tokamak power plants. The localized power and particle fluxes impacting the plasma-facing components risk exceeding current technological limits, particularly regarding the power-handling capacity and materials' erosion lifetime. Modeling the plasma boundary is key in evaluating and optimizing magnetic topology for plasma exhaust management, identifying constraints and parameters for the wall and divertor design. One of the objectives of the experimental campaign for the SPARC tokamak[1] is to identify high-dissipation divertor scenarios suitable for the ARC power plant[2]. A dedicated divertor chamber has been integrated into the SPARC design to experimentally test the promising alternative divertor configuration of the X-Point Target (XPT). This magnetic configuration integrates a long-legged divertor and a secondary X-point within the divertor volume. These features offer significant advantages, including the localization of the peak radiation in the divertor region and an increase in the plasma-wetted area on the divertor targets. Experiments and modeling[3,4] have shown a significant reduction in the heat flux peak on the wall compared to conventional divertor configurations, achieving a fully detached divertor with corresponding cold targets (Te≲5 eV).
In the present study, the plasma conditions are characterized using the edge transport fluid code SOLEDGE3X[5] in 2D, integrated with the kinetic code EIRENE for the neutral particle description. The modeling domain extends up to the first wall, providing power and particle flux distributions in the Scrape-Off Layer (SOL), the divertor region, and the entire wall perimeter. Two configurations of possible XPT experiments for SPARC are studied: the Connected and the Disconnected Double Null (CDN/DDN). The power exhaust performance of these scenarios is compared against each other and as well as against the CDN with vertical divertor targets, one of the conventional configurations planned for SPARC. All the cases are in H-mode, with the toroidal field of 12T and plasma current of 5.0 MA, at the same upstream separatrix density condition, n=0.5x1020m-3, and with the same power entering the SOL, PSOL=17MW. Extrinsic divertor impurities are excluded at this stage, considering a pure deuterium plasma. The modeling shows the differences between the three cases in particle, power, and radiation distributions in the SOL, as differences in fluxes to the wall. This detailed 2D study provides a starting point for assessing the potential of the XPT configuration as a high-dissipation regime for a future ARC power plant.

Acknowledgments: This work is supported by Commonwealth Fusion Systems.
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Design Progress Toward the ARC Fusion Power Plant

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Monday Parallel 3a - Next Steps I, Kresge Main Theater (Building W16, upstairs), June 23, 2025, 4:00 PM - 5:30 PM

The ARC fusion power plant will produce 400 MW net electric and is targeting operation in the early 2030s at a site in Chesterfield County, Virginia. ARC will build upon lessons learned from SPARC, focusing on a high-field, standard aspect ratio, ion cyclotron-heated, inductive, DT tokamak. ARC design is progressing in a series of rapid iterations, incorporating inputs from plasma physics, mechanical engineering, process engineering, materials science, nuclear engineering, architecture, and other disciplines. Critically, ARC is being designed with capital cost, levelized cost of electricity, and customer needs as the central drivers from the beginning, ensuring that the design process results in a product which is economical and fits market needs. The current iterative design effort will culminate in a pre-conceptual design for ARC and an accompanying plasma physics basis. This pre-conceptual design will include key flexibility to incorporate learnings from early SPARC operation and other parallel R&D efforts, such as on the molten FLiBe salt blanket, plasma facing and structural materials, remote maintenance of tokamak components, tritium technology, and other areas. This presentation will describe the current state of the ARC design, ongoing design efforts, and connections to key R&D inputs.

STEP update: Learning lessons through an evolving concept design

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Monday Parallel 3a - Next Steps I, Kresge Main Theater (Building W16, upstairs), June 23, 2025, 4:00 PM - 5:30 PM

The Spherical Tokamak for Energy Production (STEP) programme aims to deliver a UK prototype fusion energy plant, targeting 2040, and a path to commercial viability of fusion. Fusion plant design is an unprecedented engineering challenge that requires extrapolation into uncharted territory. Its complex and evolving nature, often arising from unpredictable behaviours, drives change across the system of systems and lifecycle, making design inherently difficult. Furthermore, as we push the boundaries, new scientific and technological discoveries will emerge, likely leading to a redefinition of fundamental rules and requiring an adaptive approach to engineering design.
A concept for the STEP prototype powerplant (SPP) was published [1], which highlighted the challenges of creating a balanced system of systems and translating complex integration challenges into practical engineering and design. This led to a better understanding of the design defining features, key risks and uncertainties, and how these relate to the design objectives of generating 100 MWe of net electrical power, be tritium self-sufficient, demonstrate high-grade heat, and demonstrate a route to commercial levels of plant availability. By reviewing the concept against the objectives, the design team has also explored a larger machine size, architectures and a solid breeder in the blanket. Technology development roadmaps have been developed that increase design confidence by identifying the key testing and prototyping capabilities that will be required. Alongside these, the overall programme will also make strong use of digital modelling to accelerate design development and provide enhanced predictive capability. In particular, the aim is to (a) provide a rigorous basis for extrapolation from the conditions of an experimental test rig to those of SPP, and (b) to understand the interfaces and couplings between the different subsystems. In this contribution, we will illustrate these features through an updated STEP design, highlighting the key design changes that have been made and their wider impact. We will then discuss the key technology development plans going forward, designed to target the minimisation of risk and uncertainty.

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Preparation of the European Fusion Roadmap

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Monday Parallel 3a - Next Steps I, Kresge Main Theater (Building W16, upstairs), June 23, 2025, 4:00 PM - 5:30 PM

The design of DEMO in Europe benefits largely from the experience gained from the design, licensing, and construction of ITER, which remains the crucial machine for the validation of the DEMO physics and part of the technology basis. However, work done in the pre-concept design phase and the gate review revealed that the DEMO design and operating space is heavily constrained by physics and technology with key technological challenges beyond ITER. Several studies were recently launched by the DEMO Central Team (DCT) to define the optimum DEMO design space heavily constrained by physics and technology, aiming to minimize either the machine size or the technical risks, and to facilitate an earlier DEMO deployment.
The reduction of the electricity output (now set to about 350 MWe for DEMO in Europe, still within the original stakeholder requirement range) does not bring any significant size reduction if the other key stakeholder requirement of a 2-hour pulse duration is maintained. However an adequate plasma safety factor margin is now introduced, which provides operational safety (less disruptivity) for the plasma scenario, and allow to “insulate” the engineering design from physics uncertainties, coming from extrapolation to present to larger future devices, leveraging the performances on such plasma safety factor. High-field magnets do not lead to a reduction in machine size, as large structures are needed to withstand the forces, and the limiting factor remains the transient heat load protection criteria for the divertor, based on ITER-like technology. We see significant benefits going to a DEMO with a lower aspect ratio both from the standpoint of minimizing the heat load on the divertor, the size of the toroidal field coils mechanical structures and adding adequate plasma stability margin. These tasks integrate initial inputs available for design and integration activities, including simulations and validations, CAD models, load assessments, and sensitivity analyses accounting for uncertainties. A dedicated working group has been established to explore potential improvements to the DEMO design point using lower Technical Readiness Level (TRL) solutions. These include advanced structural materials, innovative plasma configurations, alternative machine architectures for maintenance strategies, and alternative neutron shielding materials. In addition, essential enabling technologies (breeding blanket, T fuel cycle, divertor, materials, remote maintenance) were found to have a very low maturity. To mitigate this low TRL, the feasibility assessment of a plasma-based VNS has been undertaken by the DCT and some very preliminary design considerations are provided here. The VNS would serve as a facility for testing and qualifying crucial nuclear fusion technology components in parallel to DEMO design and construction. This facility would be complementary to both ITER, which is focused on burning plasma physics, and to DONES, which is focused on large dpa in small material samples. The VNS feasibility study was concluded with the basic design of the tokamak machine integrated with the main plant systems including provisions for maintenance and able to meet safety & licensing requirements. VNS offers the relevant environment to test and qualify fusion nuclear components and a large area to install BB TBMs i.e. ≈25 m^2 on the outboard side. The EUROfusion Scientific and Technical Advisory Committee (STAC) reviewed the outcome of the VNS feasibility study.

Characterization of Neutron Flux Monitors of the SPARC Tokamak

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Monday Parallel 3b Diagnostics, Instrumentation, Data Acquisition & Management I, Sala de Puerto Rico (Building W20 Room 202), June 23, 2025, 4:00 PM - 5:30 PM

In the SPARC tokamak [1], neutron flux monitors (NFM) will measure real-time neutron flux in the tokamak hall and convert the measurement to fusion power via calibration. Such measurements are of great importance for demonstrating fusion power breakeven (Q = Pfus/Pin > 1), machine protection and plasma operations. To cover the wide dynamic range of DT and DD neutrons in different SPARC campaigns [2], from ~10^8 n/s calibration range to 5×10^{19} n/s primary reference discharge (PRD), multiple types of NFMs are proposed [3]. This work will present the latest results of characterization of a BF_3-filled proportional counter (PC) and a U238 fission chamber (FC). BF_3 PC is a highly sensitive detector for calibration range. U238 FC, which has overall low sensitivity and is especially insensitive to neutrons < 1 MeV, is a promising candidate for direct fusion neutron measurements for PRD. Experiments are performed to characterize these detectors' linearity, gamma discrimination capability, sensitivity to source neutrons of different energy, and flux shaper design. DT and DD neutron generators are operated at different voltages to produce neutrons of different energy and intensity. OpenMC[4] is used to simulate both the experiments and the detectors' performance in SPARC. The U238 FC is equipped with a Borated polyethylene (B-PE) box with an opening channel to filter out scattered neutrons from the surroundings so as to prioritize direct neutrons [5][6]. The FC showed great linearity to both DT and DD neutrons and insensitivity to gamma. BF_3 PC, as a thermal neutron detector intrinsically, will have both a B-PE box to prioritize direct neutrons and a high density PE (HDPE) sleeve to moderate fast neutrons locally. More experimental results will be available before the conference. We also considered a Lithium-based scintillator as a potential upgrade to the NFM system [7]. This detector also showed great linearity to neutrons and decent gamma discrimination performance.

Acknowledgement:
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Status and Plans for SPARC's Early Campaign Diagnostics

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Monday Parallel 3b Diagnostics, Instrumentation, Data Acquisition & Management I, Sala de Puerto Rico (Building W20 Room 202), June 23, 2025, 4:00 PM - 5:30 PM

SPARC is a compact, R=1.85, high field, Bt0=12.1 T, tokamak presently under-construction in Devens, MA. It will be used to demonstrate net fusion energy, Qfus > 1, for the first time in a magnetically confined plasma. Achieving this milestone in its first campaign using a DT, ICRF-heated L-mode requires a range of diagnostics for basic plasma control, control at high current and power density, tools for scientific learning and accurate measurement of performance metrics like fusion power. Specifics of the ~50 classes of diagnostics [1] are outlined, including how they are engineered into the overall facility and the tokamak's port-plugs, plasma facing components and vacuum vessel. A summary of in-progress fabrication and assembly activities is included, highlighting opportunities where enhancements are possible using private-public partnerships. To the extent possible, a common hardware platform for data acquisition and control has been developed to simplify integration, operations, and software development. The platform supports 96 channels of analog to digital sampling at 18 bits and 10 MS/s, and enables onboard real-time digital signal processing. This platform is intended to be used across many diagnostics, with limited remaining gaps filled with the small range of COTS products. An overview of how data from SPARC's diagnostics will be organized and maintained to facilitate fast, inter-pulse learning is provided.

This work supported by Commonwealth Fusion Systems

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Broadband Millimeter-Wave Measurements with ITER Prototype Martin-Puplett Interferometer

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Monday Parallel 3b Diagnostics, Instrumentation, Data Acquisition & Management I, Sala de Puerto Rico (Building W20 Room 202), June 23, 2025, 4:00 PM - 5:30 PM

Over decades, Fourier Transform Spectrometers (FTSs) have become a standard diagnostic tool to measure Electron Cyclotron Emission (ECE) spectra in the millimeter-wave range in leading fusion devices like Alcator C-Mod [1], DIII-D [2], EAST [3], JET [4] [5], LHD [6], JT-60 [7], JT-60U [8], TFTR [9] , and W7-X [10] . FTSs provide critical information about radiative power loss, non-thermal electron populations, and global temporal and spatial characteristics of the electron temperature profile Te. Due to its high accuracy and reliability, FTSs are indispensable for cross-calibrating other diagnostic instruments, such as heterodyne radiometers with multiple active components. Furthermore, the consistency between absolutely calibrated FTSs and Thomson scattering systems can be tracked by comparing the Te profiles determined by the aforementioned diagnostics. This is a prerequisite of any non-thermal study which requires to measure ECE radiation of perpendicular polarisation directions over multiple harmonic ranges for the radial as well as the oblique line of sights. For the above reasons, two FTSs of Martin-Puplett interferometer type [11] will be used at ITER, to probe ECE spectra, ranging from 70 GHz to 1 THz. Both FTSs probe simultaneously ordinary and extra-ordinary polarizations of the ECE radiation for either perpendicular or oblique (9.25°) view on the ITER plasma. ITER's FTSs are designed for high throughput, excellent polarization sensitivity, and ease of calibration. In addition, the signal loss is reduced leading to a precise spectral analysis, making the FTSs ideal for ECE studies on larger time scales.
A prototype FTS system has been [12] [13] built with a 15 mm double-sided scanning range, operating at 50 Hz. This setup achieves a spectral resolution of 10 GHz and a temporal resolution of 10 ms. It serves as a testing platform for key ITER components, including smooth-walled circular (ID 72mm and OD 88mm) waveguides[14], Miter bends and vacuum compatible waveguide joints designed for broadband transmission lines and local blackbody calibration sources. The performance of these transmission lines, optimized for low-loss signal transport over a 40 to 50 meters distance, has been measured using a newly developed electrically heated calibration source. The average transmission coefficients for a ~10 m transmission line, excluding water vapor absorption bands, were determined to be 0.64 for the frequency range 100-400 GHz and 0.43 for 400-1000 GHz. The hot calibration source features a pyramidal silicon carbide surface (200 mm in diameter), a spiral heating element, and a precision temperature control system, ensuring a stable and reliable calibration procedure for the diagnostic at hand.
This paper highlights the design and current status of the FTS diagnostic hardware, calibration setup, data analysis techniques, and system characterization. Additionally, recent ITER prototype transmission measurements conducted using the hot/cold technique are discussed.

The TOFU synthetic diagnostic code suite applied to the SPARC X-ray crystal spectrometers: throughput, spectral fitting, and tomography

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Monday Parallel 3b Diagnostics, Instrumentation, Data Acquisition & Management I, Sala de Puerto Rico (Building W20 Room 202), June 23, 2025, 4:00 PM - 5:30 PM

High-resolution X-ray crystal spectroscopy (XRCS) has been a workhorse on numerous tokamaks to measure the ion temperature and toroidal rotation profiles from the Doppler broadening and shift, respectively, of intrinsic or seeded impurity line radiation emission. SPARC will be a first-of-its-kind tokamak that will similarly employ this diagnostic. The TOFU open-source, python-based synthetic diagnostic code suite was originally developed to analyze measured XRCS spectra from the WEST tokamak [1]. Here we applied the code as a predictive tool as we designed the XRCS system for SPARC [2, 3].

In this talk we will present cross-validation with the similar, but foundationally different XICSRT code suite developed and already benchmarked for the XRCS system on ITER [4]. Excellent agreement is found across codes in both the shape and magnitude of the resultant photon flux on the detector for spherically, cylindrically, conically, and toroidally bent crystals in rotating plasmas with radially peaked or hollow emissivity profiles. Rocking curves are calculated by TOFU for the quartz and germanium crystals considered at arbitrary Miller indices including thermal expansion and miscuts using ideal dynamical diffraction theory and were benchmarked against pyTTE [5]. We then use the companion code, SPECTRALLY, to fit the detector images using Gaussian and pseudo-Voigt line shapes as well as chi-squared or Cash statistics. TOFU is then used to perform Phillips-Tikhonov inversion-regularization [6] of the radial emissivity profiles, and the error in reconstructed local ion temperature and toroidal rotation profiles given a certain number of installed lines-of-sight into the plasma is discussed.

Work supported by Commonwealth Fusion Systems.

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Conceptual Design and Layout of the Eos Power System

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Monday Parallel 3c - Power Supply Systems, Kresge Little Theater (Building W16, downstairs), June 23, 2025, 4:00 PM - 5:30 PM

Thea Energy is pursuing a roadmap toward a fusion pilot plant through the development of Eos, a sub-breakeven, deuterium–deuterium, beam-target fusion stellarator neutron source facility. Eos is designed to produce radioisotopes, and by incorporating a tritium breeding blanket, it will generate up to 0.2 grams of tritium per day or approximately 70 grams per year. Eos requires about 40 MW of electric power to support this ambitious goal to achieve a neutron production rate of 2.5 × 10^17 neutrons per second.

A primary objective of the Eos stellarator project is to advance new stellarator technologies by demonstrating several key milestones. These include achieving plasma confinement times longer than 400ms sustaining 200 million Kelvin plasma temperatures and maintaining high-performance operational regimes. Ultimately, the project aims to showcase an integrated system capable of breeding tritium using specialized absorbers in a dedicated blanket system, thereby moving closer to realizing a scalable, reliable fusion power plant. This vision aligns with Thea Energy’s broader mission of contributing to a clean, carbon-free energy infrastructure.

The Eos stellarator design features a unique combination of superconducting Encircling Coils and modular Shaping Coils, which precisely form magnetic field geometry to maintain stable plasma confinement. Meanwhile, advanced heating and current drive systems, such as Electron Cyclotron Resonance Heating (ECRH) and Negative Neutral Beam Injection (NNBI), will support the high temperatures required for effective D-D fusion reactions. By integrating these technologies, the Eos stellarator will serve as a testbed for both plasma physics research and engineering innovations that directly inform the path toward a future pilot plant.

This paper focuses on the conceptual design and layout of the Eos Power Supply System, which delivers critical power to each of the major subsystems: Encircling Coils, Shaping Coils, ECRH, and NNBI. The overarching power architecture was driven by physics requirements and system-level constraints, ensuring that each subsystem operates under optimal conditions. To achieve this, novel topologies of power supplies and control algorithms have been developed and validated using extensive simulations that capture the variety of operating modes Eos will experience. These simulations were performed using high-fidelity modeling tools to account for transients, load variations, and real-time control needs.

A key outcome of this work is the establishment of a comprehensive power consumption profile for Eos, detailing both peak and average loads. This profile underpins the conceptual design of a local distribution substation, enabling the power system to seamlessly handle significant load fluctuations. By mapping these loads to the broader facility infrastructure, Thea Energy ensures that the Eos facility remains reliable, efficient, and adaptable to evolving experimental scenarios.

Lastly, the conceptual layout of the Eos Power Supply System was proposed, which integrates a power distribution system and all power supply system components in a manner that supports rapid testing, iterative development, and straightforward scalability. This approach not only facilitates smooth day-to-day operations but also lays a robust foundation for Thea Energy’s long-term objective: developing a commercial-scale fusion pilot plant.

Electrical Energy Storage for STEP – Requirements Capture and Understanding the Technology Landscape

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Monday Parallel 3c - Power Supply Systems, Kresge Little Theater (Building W16, downstairs), June 23, 2025, 4:00 PM - 5:30 PM

STEP (Spherical Tokamak for Energy Production) is an ambitious program with the goal of designing and building a prototype fusion energy plant in West Burton (Nottinghamshire) that can demonstrate the delivery of net electricity to the electrical grid. Its ultimate goal is to pave the way for the commercial viability of fusion technology.

Fusion power plants, like STEP, have an array of large parasitic electrical loads, which are necessary to initiate and sustain the fusion reactions. These loads are typically much larger and more dynamic than the ones found on conventional power plants. During start-up mode and throughout the experimental/ commissioning phases of STEP, these parasitic loads need to be powered by external sources, as STEP will not be capable of power generation during these periods. STEP is planning to supply these loads using power sourced from the electricity transmission grid and/or some form of on-site Electrical Energy Storage System. The exact split between the two power sources will be dependent on the future electrical characteristics of the 400 kV grid connection at the West Burton site.

STEP’s electrical distribution network is expected to experience periods in which rapid electrical loading and unloading will occur. These power dynamics are caused by the large number of power converters embedded within the electrical plant, which support plasma control within the tokamak. The majority of these fast power transients are expected to be managed by Electrical Energy Storage Systems. This is necessary in order not to breach the limits and constraints imposed upon STEP by the respective Industry Codes (e.g. Grid Code). In the future, it may be beneficial to dedicate some Electrical Energy Storage Systems to maintaining plant voltage levels within acceptable limits and to providing reactive power support across a wide range of load dynamics. Furthermore, like in conventional power stations, Electrical Energy Storage Systems could provide back-up power during emergency events.

In the wider context of low carbon energy systems, there is great interest in Electrical Energy Storage for grid applications. This is due to their ability to support variable renewable energy sources that are connected to the electricity grid in a decentralised way. The focus is mainly on battery energy storage systems (BESS), due to their large energy density compared to other technologies. But STEP, as well as other fusion power plants, have a wide range of ESS requirements, and for some of them, energy density may not be the most important characteristic.

This work presents STEP’s Electrical Energy Storage requirements and explores the current and future technology landscapes, considering the solutions that are best suited for the support of STEP’s future operations.

Development of RF Solid-State Amplifiers for ITER HNB ion sources

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Monday Parallel 3c - Power Supply Systems, Kresge Little Theater (Building W16, downstairs), June 23, 2025, 4:00 PM - 5:30 PM

SPIDER is the full-scale prototype of the ion source for the ITER Heating Neutral Beam Injector, while MITICA serves as the full-scale prototype of the ITER Heating Neutral Beam Injector itself. The development and operation of SPIDER and MITICA, along with the transfer of experience gained on the ITER injectors, are part of the collaboration between Consorzio RFX and the ITER Organization.
The ITER Heating Neutral Beam (HNB) Injectors produce negative ions using plasma generated by 8 RF antennas, known as RF drivers, supplied in pairs by 4 RF generators. Originally, free-running tetrode oscillators were used, but significant limitations and issues exhibited from the very beginning of SPIDER operation pushed for the change of the technology to solid state switching amplifiers. Positive results from tests conducted in facilities like BATMAN and ELISE at the Max Planck Institute for Plasma Physics in Garching have reinforced this decision.
The development of RF Solid-State Amplifiers (RFSSA) rated for 200 kW on 50 Ω, operating at 1 MHz ±80 kHz, was addressed to meet the specific requirements of the HNB: the management of resonant loads, such as the drivers and their matching network, as well as plasma initiation and fault tolerance in cases of plasma loss or RF breakdowns.
The RFSSA implements 24 RF modules rated at 10 kW peak, combined through a ferrite core multi-primary transformer. This choice allows full power operation either in moderately mismatched conditions or with a faulty module, allowing for improved operational margins and plant availability. The base module includes an independently controlled DC link power supply, a full H-bridge inverter driven at the desired frequency, and a two-cell output filter. The RF output power level can be controlled either with the DC link voltage for slow power transients, of through the H-bridge PWM for fast variations.
Integration of the RFSSA units into the SPIDER and MITICA plants was designed taking into account the lessons learned from the past experience with RF oscillators: implementation of countermeasures for RF stray currents, adoption of design margins for the mutual coupling between RF circuits, as well as requirements for the filters to limit the voltage total harmonic distortion.
Key innovations in the amplifier control design include automatic recovery mechanisms following a breakdown that involves the RF circuits in the source, and the capability for adjustable phase frequency-synchronized operation of the RFSSA supplying different driver pairs to mitigate beat phenomena observed with oscillators.
Moreover, to minimize the integrated commissioning time on site, factory testing was planned under conditions that closely simulate operational scenarios, by means of a resonant dummy load that mimic the actual ion source load.
The RFSSA development, awarded by Consorzio RFX to the company OCEM Energy Technology, has now reached completion, with all systems tested and installed, ready for operational deployment in the SPIDER and MITICA experiments. The detailed design, the addressed issues, the results of the testing campaigns and the performance of the amplifiers will be presented, confirming the effectiveness of the design choices made throughout the project.

Neutron Detector and Generator Characterization for the BABY Experiment

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Monday Posters 3, Lobdell (Building W20 Room 208), June 23, 2025, 4:00 PM - 5:30 PM

The Liquid Immersion Blanket: Robust Accountancy (LIBRA) project explores the liquid immersion blanket concept as a tritium breeding fusion reactor blanket design. This project addresses the need for tritium sustainability, which is critical for the viability of DT fusion reactors. Before the full scale LIBRA system is developed, operations at a smaller scale are underway in the Build A Better Yield (BABY) Blanket campaign, which is performing irradiations of 1L of molten ClLiF salt using 14MeV DT fusion neutrons produced by a neutron generator. In order to make assessments of the efficacy of the tritium breeding system, the neutron input received by the molten salt must be well understood. Neutron energy and flux are measured using a Cividec Diamond DT-Fusion Neutron Monitor, a Cividec Diamond Proton Recoil Telescope, and activation foils. The calibration of these detectors and the design of the measurement setup will be presented in this poster. Additionally, the neutron generator has been characterized through a series of measurements, which will be presented alongside their implications for the BABY experiments. This work allows for a more complete understanding of the neutron environment of the BABY experiments, helping to evaluate the effectiveness of the tritium breeding design.

Quality Risk Identification And Process Optimisation Using Stochastic Petri Net And Semi-Markov Process: A Probabilistic-Driven Approach

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Monday Posters 3, Lobdell (Building W20 Room 208), June 23, 2025, 4:00 PM - 5:30 PM

Stochastic Petri Net (SPN) and Semi-Markov Process (SMP) are powerful and widely used approaches for modelling, dead-lock detection and safety analysis of stochastic processes in complex systems. However, applying these two methods is rarely seen in formal risk analysis, especially for quality risks (QRs), which refer to the system's capability and performance to fulfil its objectives, measured in the time domain. This paper presents a novel approach to identifying QRs during the conceptual design stages based on the fundamentals of SPN and SMP, using either statistical analytics or Monte Carlo simulations to solve the design problem, depending on the complexity of the SPN network. The framework provides a quantitative approach to facilitate hardware configuration selection with the cost constraints between different design concepts and benchmarking them under the Cumulative Distribution Function (CDF). Two case studies are presented to demonstrate the method's applicability: the design of an industrial AGV and the concept development of a remote maintenance system for the In-Bioshield area of the DEMO fusion power plant. Initial results showed potential in identifying quality risks, addressing the systems' worst-case scenarios, and finding optimal design specifications in the early design stages.

Developing a Safety-in-Design Methodology for a Fusion Safety and Reliability Case

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Monday Posters 3, Lobdell (Building W20 Room 208), June 23, 2025, 4:00 PM - 5:30 PM

As strides are being made towards overcoming the physical and engineering hurdles to fusion power, attention must also be paid to issues relevant to successful commercialization. Central to addressing such issues is the incremental development of a safety and reliability case that appropriately identifies and evaluates risk in a fusion power plant design. Although fusion is to be regulated by the nuclear regulator in the United States, its risk profile differs significantly form that of previously regulated fission reactors. Thus, the approach to developing a safety and reliability case would benefit from a basis in familiar risk assessment and regulatory practices, while embracing a risk-informed approach not tethered to that of the nuclear fission industry. To this end, we report on our progress in applying the Safety-in-Design (SiD) methodology to fusion plant design and safety & reliability analysis.

SiD is a systems engineering-based approach that uses established industry assessment methods, both qualitative and semiquantitative, to provide a technology neutral approach for designs with novel technologies and limited operations experience. Decomposing a proposed design using functional and physical architectures facilitates a graded, risk-informed analysis that ensures depth of analysis is appropriate to a given system. For example, although prior experience from fission designs might suggest prescriptive use of highly quantitative methods like Probabilistic Risk Assessment (PRA), this may not be necessary given the substantially differing hazard profile of present fusion designs. The SiD methodology presents two key benefits. First, it can help uncover and resolve safety and operability issues early in the design process, reducing financial and time costs incurred when uncovering such issues later, while improving safety and reliability outcomes. Second, a robust, SiD-based safety case aids in communication with regulators, investors, and other stakeholders and—thanks to the use of internationally-recognized risk assessment and systems engineering methods—can also facilitate multinational collaboration.

This paper discusses the approach and progress in tailoring an SiD approach for fusion applications. Three primary avenues of research are presented. First, the finalization of a body of knowledge (BoK) compiles and curates the state of the art in fusion designs and their technology, focusing on associated hazards, reliability issues, regulations, and their management strategies as they support a systems engineering approach. Second, preliminary applications of SiD to extant fusion designs—including a post-hoc analysis of the Joint European Torus's (JET) Active Gas Handling System (AGHS)—are discussed. Finally, we discuss the potential to incorporate Model-Based Systems Engineering (MBSE) into the SiD methodology.

Fabrication Experience and Requirement Comparison of 316L(N)-IG Forging for ITER In-vessel Components

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Monday Posters 3, Lobdell (Building W20 Room 208), June 23, 2025, 4:00 PM - 5:30 PM

The austenitic stainless steel 316L(N)-IG is mainly employed for the structural material of the core machine considered for the in-vessel components for the ITER. Based on in-service experience and research results from fusion programs, a solution-annealed 316L steel type proves most suitable material to resist a high radiation dose. For application in ITER, minor modifications are necessary, particularly concerning the Co (cobalt) and Nb (niobium) contents, to align with radiological safety limits and with the rewelding requirement. In the case of fusion reactor, activated cobalt significantly influences occupational dose levels during maintenance and severe accidents. Additionally, niobium produces long-lived radioisotopes that could play a crucial role in decommissioning and waste disposal for in-vessel components.
The VQC 1A components of ITER in-vessel components shall be made using 316L(N)-IG. Specially, all VQC 1A in-vessel components with a final thickness less 5 mm should be fabricated from cross-forged (and/or upsetting) material, which can be either Electro-Slag Re-melted (ESR) or Vacuum Arc Re-melted (VAR).
This paper describes the entire fabrication process and requirements of the 316L(N)-IG for the ITER VQC 1A components in terms of RCC-MR 2007. The present work focuses on the inspection method and evaluation level of Ultrasonic Test (UT) to verify the quality of forging material. From the conservative point of view, the conventional standards of ASME Section V, Article 23, SA-745 (ASTM A745) and EN 10228-4 for the Austenitic Stainless Steel, were also compared.

Electromechanical analyses comparison of different supporting structures for the DTT ICRH antenna

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Monday Posters 3, Lobdell (Building W20 Room 208), June 23, 2025, 4:00 PM - 5:30 PM

The Ion Cyclotron Resonance Heating (ICRH) antenna of the Divertor Tokamak Test (DTT) facility is expected to perform several tasks through the radio frequency waves injected into the plasma core. As the power coupling decreases with the distance from the plasma edge, an in-vessel frontal part facing the plasma directly is necessary to reach the required power. The antenna can therefore undergo demanding electromagnetic (EM) loads (i.e. the Lorentz force density J x B) due to the interaction of the total magnetic field B with the induced current density J, caused by rapid plasma current and position variations in case of a disruption event. Considering a major disruption event with a current quench time of 4 ms starting from the DTT reference single null equilibrium, the present work wants to focus on the eddy currents pattern and the consequent forces and torques acting on and transmitted to the cantilevered supporting structure sustaining the in-vessel frontal part. The hypothesis of infinitely rigid body is made for the frontal part components to transport the resultant forces and torques. Moreover, two different architectures of the supporting structure are compared. The eddy currents computation and the structural analyses are performed by the means of the tools CARIDDI and ANSYS structural respectively.

Keywords: Divertor Tokamak Test, ICRH antenna, supporting structure, structural analysis, electromagnetic loads

DEMO divertor target mock-ups with tungsten fiber reinforced tungsten monoblocks

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Monday Posters 3, Lobdell (Building W20 Room 208), June 23, 2025, 4:00 PM - 5:30 PM

Within the EUROfusion Consortium, in the context of the workpackage Materials (WPMAT), tungsten fiber reinforced tungsten composite (Wf/W) materials have been developed to overcome the intrinsic brittleness of pure W as potential plasma facing material (PFM). Wf/W materials have showed improved fracture toughness and increased ductility, reducing crack propagation by exhibiting a pseudo-ductile behaviour. The “pseudo ductility” is generated by the interaction between the W fibers and the W matrix by means of energy-dissipation mechanisms.
In this context, Wf/W materials have been used to realize small-scale mock-ups based on the so-called ITER-like monoblock design. Among the different type of Wf/W materials, long-fibers Wf/Ws have showed improved strength and reinforcement effect compared to short-fiber Wf/Ws. At the Special Technologies Laboratories of ENEA Frascati (ENEA-TES), Wf/W monoblocks with different characteristics have been joined to CuCrZr ITER Grade (CuCrZr-IG) cooling pipes by hot radial pressing (HRP) with an Oxygen Free Electronic Grade copper (OFE-Cu, Cu) interlayer manufactured by casting. In order to assess the quality of the fabrication processes, Non-destructive examination (NDE) by ultrasonic testing (UT) have been performed. Long-fibers Wf/W monoblocks can easier replace pure-W monoblocks in the current consolidated fabrication process of a target divertor for ITER which foresees a Cu interlayer and diffusion bonding joining with a CuCrZr-IG cooling pipe.

MHD flows in the lower expansion chamber of a generic outboard blanket segment for DEMO

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Monday Posters 3, Lobdell (Building W20 Room 208), June 23, 2025, 4:00 PM - 5:30 PM

The water-cooled lead lithium (WCLL) blanket has been chosen as European reference concept for a DEMO reactor. In the proposed design, liquid metal velocities in the columnar-arranged breeder units (BUs) are very small. This weak purge flow is required to circulate lead lithium (PbLi) towards external ancillary facilities for purification and tritium extraction. When the electrically conducting PbLi moves through the strong plasma-confining magnetic field, magnetohydrodynamics (MHD) affects the flow distribution and creates increased pressure drop compared to hydrodynamic conditions.

Previous MHD analyses have been performed for a single column of breeder units with focus on pressure drop in poloidal manifolds that distribute PbLi among BUs. Manifolds have been identified as key and possibly critical components for the performance of blanket modules, since the major fraction of pressure drop arises in these elements. Moreover, they determine partitioning of the PbLi flow among breeding zones.

While the ITER TBM is formed only by two columns of BUs, in a DEMO reactor design the number of parallel poloidal feeding and draining manifolds is larger. As a consequence, liquid metal has to be distributed from a single supply line into those channels via complex 3D geometries. In the present design of a blanket outboard segment, this is achieved by a toroidal expansion chamber located in the lower part of the module. The present analysis considers the MHD flow in the chamber, that expands from a radial circular entrance pipe along the toroidal direction towards two circular holes in the back plate through which the liquid metal is then distributed into the poloidal manifolds. Pressure and electric potential distributions are calculated via an asymptotic numerical approach valid for intense magnetic fields when inertia effects are negligible. Pressure losses, which arise preferentially from 3D MHD effects in the expansion and contraction regions, are quantified.

R&D activities carried out for cryopumping to provide customized needs for Fusion Research

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To cater the need of future energy demand nuclear fusion is one of the promising solution. The readiness of the fusion power is strongly depends on the availability of its subsystems and their associated technologies. Cryopumping is one such technology and addressed to provide the solution to the vacuum. Various tokamaks like JET, ASDEX, EAST, DIII-D, JT60 has considered cryopumping to get the clean vacuum either in-vessel or out-vessel application and is also considered for ITER. Aiming to the need for Indian fusion program, R&D activities are being carried out at Institute for Plasma Research (IPR) to develop cryopump for the various customized needs. The major areas are selection of sorbent and adhesive materials and their thermal and vacuum characterization, design and development of hydroformed cryopanels and shields, pump design, prototyping and test experiments for vacuum pumping performances. For the sorbent characterization BET pore and adsorption isotherms were studied for different sorbent types at 77 K to 4.2 K. Cryogenic adhesives were developed with industry and tested. Hydroformed cryopanels were designed and fabricated in sizes 500 mm (L) x 100 mm (W) x 1.5 mm (sheet thickness) for small scale cryopumping experiments and 1000 mm (L) x 200 mm (W) x 0.8 mm (sheet thickness) for single and multi-panel cryopumping experiments. Pump geometries were optimized using MOLFLOW+ software developed by CERN for better transmission probabilities and capture co-efficient estimation. For Small scale to mutipanel cryopump arrangements ~ 2,000 l/s to more than 30,000 l/s pumping speeds were achieved for helium gas when the cryopanels were operated near 4.5 K using liquid helium. As a part of the performance evaluation cryopanels were cooled by liquid nitrogen (at 80 K) for leak tightness and structural integrity. During the process, pumping experiments were carried out for the pumping of nitrogen and argon gas and significant pumping effect is gained. Inspiring by this, for the immediate application customized liquid nitrogen based cryopump developments were initiated. First prototype 80 K cryopump with a pump opening 400 mm provide pumping speed of 3616 l/s (Experimental average) for Nitrogen and 15928 l/s for water vapor with tested pumping capacity more than 6000 mbar-l/s. One such 80K pump is custom built and is used in SST-1 tokamak during its baking cycle for the pumping of water vapor and nitrogen having pumping speed greater than 25,000 l/s for H2O and 3200 l/s (Max.) for N2 respectively. The pump has been operated when the in vessel components are at ~150 °C making it one of its first kind. This shows large potential for the cryopump operating at 80K. For the applicability one small size similar kind 80K cryopump has also been integrated to the IPR's High Heat Flux (HHF) Test Facility and experiments were performed. The pump provides 1197 l/s pumping speed for nitrogen and 5106 l/s for water vapor. In this paper, in house research and developed activities carried out at IPR in the diverse field of cryopumping and its application experience will be presented.

Elastic force compensation via negative stiffness mechanism for the ECRH steering launcher of DTT

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The full power configuration of the Divertor Tokamak Test (DTT) facility will include 32 independent Electron Cyclotron Resonance Heating (ECRH) front-steering launching mirrors. A highly compact, 2-degrees-of-freedom steering mechanism based on in-vessel piezoelectric walking drives is currently under design. This solution is intended to minimize space occupation within the ports – allowing for the launchers within the limited DTT duct space – while optimizing dynamic performance and control bandwidth. Wherever feasible, flexures replace traditional hinges, with the combined advantages of eliminating wear and backlash, thus extending component lifespan and enhancing steering accuracy. At the same time, flexible joints introduce elastic resistance to the steering motion, which must be counteracted by the actuators. This reduces the force available for resisting other external disturbances, like electromagnetic loads, and decreases the maximum acceleration achievable by the steering mirror, which negatively affects the promptness to control input changes. In order to strongly mitigate elastic resistances, a negative stiffness element called "stiffness compensator", based on pre-compressed springs, has been designed. A detailed analysis of the stiffness compensator is presented, along with a robustness assessment taking into account manufacturing uncertainties. The result is a "quasi-zero-stiffness" (or "statically balanced") steering mechanism, which can benefit from the increased positioning accuracy following from the absence of friction without compromises on the available driving force.

Background oriented schlieren velocimetry of helium coolant flow in additively manufactured flow channels

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High-pressure helium gas cooling is an attractive solution for thermal management of the fusion blanket first wall, as this coolant is chemically and neutronically inert and separable from hydrogenic species. However, due to the low thermal mass of helium, geometric optimization of these channels is required to provide sufficient cooling at manageable flowrates and pumping burdens. Increasingly, analysis and optimization of these coolant channels relies on computational fluid dynamics simulations, and these require relevant experimental data for turbulence model validation.

Toward this end, a high-pressure helium gas flow visualization system has been employed to image the flow of helium in flow channels with one-sided heating, mimicking the blanket first wall environment. Flow of helium at 4 MPa pressure and flow rates up to 68 g/s (Reynolds number 60,000) is supplied to rectangular channel test sections, with uniform heating applied to the bottom wall of the channel at heat fluxes varied between 50 and 100 kW/m^2. A high-speed camera is used to image index of refraction gradients in the fluid via background oriented schlieren (BOS), and temperature and pressure instrumentation are used to characterize thermal hydraulic performance of each channel. Cross-correlation of time-resolved BOS images is then used to calculate time-averaged 2D helium velocity fields. Flow in additively manufactured channels is examined in this manner, including both featureless channels and those containing ribs and baffling as heat transfer enhancements. Key elements of the flow induced in these channels, such as separation and recirculation regions, are identified and compared to accompanying simulation predictions. Strategies are discussed for ongoing and future validation of these simulations, with the aim of model deployment for blanket cooling design and optimization.

Measurements of Fusion Performance of the Centrifugal Mirror Fusion Experiment

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The Centrifugal Mirror Fusion Experiment (CMFX) at the University of Maryland College Park is a magnetic mirror that augments plasma stability and reduces transport by using a central cathode to generate a strong radial DC electric field in order to induce a strongly sheared azimuthal ExB flow. The neutron emissions from the plasma were diagnosed with a suite of liquid organic scintillators. Measurement of neutron yield was absolutely calibrated using two well-characterized 2-inch EJ-301 and EJ-301D detectors. A more sensitive 10-inch EJ-301 detector was cross-calibrated to make time resolved measurements of the neutron rate under various experimental conditions, with peak neutron rates in excess of 10^7 n/s. Permanently installed He-3 tubes were also cross-calibrated for total yield measurements. With this suite of detectors, the fusion power and scientific energy gain of CMFX were studied as functions of bias voltage and fueling over 120 discharges. Neutron rate measurements were also used to constrain the ion temperature which is inferred to be about 1 keV according to preliminary interpretive simulations of the experiments with MCTrans++. Finally, the fusion triple product is estimated to be on the order of 10^18 keV s m-3 in the highest performing discharges.
CMFX is supported by ARPA-E Grant No. DE-AR0001270. The MIT team and equipment for this research was partially supported by Commonwealth Fusion Systems under RPP020

Towards multidisciplinary optimization for stellarator concept design

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The design of stellarator power plant concepts is challenged by the highly integrated nature of many extraordinary design problems involving numerous disciplines, leading to slow design cycles. Systems codes accelerate the design process by exploring the design space automatically using optimisation, building on 0D/1D models that are integrated to capture the complex behaviour of the full system. These codes then provide designers with a first estimate for architectural decisions, and form a starting point for further analysis by disciplinary design teams.
Unfortunately, the 3D nature of stellarators means 0D/1D models are insufficient to capture design-critical physics. Therefore, stellarator concept design requires 3D models notably for plasma and coil shaping, but also for further engineering evaluation. However in turn these currently cannot be integrated to capture the full system performance. This gap between high fidelity and high multidisciplinary design codes bottlenecks the engineering design of future stellarator power plants, such that fewer configurations can be evaluated for engineering and commercial viability.
This research addresses this issue with the realisation that this problem is not necessarily unique to fusion, reviewing state of the art concept design literature from aerospace. A qualitative comparison shows similar design challenges in aerospace, featuring complex, large-scale, and multidisciplinary optimisation problems.
Three main recommendations are extracted from aerospace literature. First, best-practice gradient computation methods are discussed, enabling efficient large-scale optimisation. Subsequently, an exploration of the field of multidisciplinary design analysis and optimisation (MDAO) shows the methods developed in aerospace enabling optimisation over heterogenous modular disciplines. The final recommendations include methods to ease (re)configuration of such MDAO systems, which may facilitate integration in fusion system engineering processes.
We argue that these aerospace recommendations would address problems currently faced in stellarator concept design, suggesting incorporation in future stellarator design frameworks. This should ultimately allow faster design of better stellarator concepts.

Multi-composition density measurement and tritium breeding analysis of LiF-LiCl and LiF-PbF2 molten salts

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Fusion power plants employing the deuterium-tritium fuel cycle will need to self-sufficiently produce tritium, due to its scarcity. This self-sufficiency may be achieved by surrounding the fusion device in a lithium-based tritium breeding blanket. Many lithium-based molten salts could be theoretically employed in such a blanket, but many of these salts lack basic physical and thermal property measurements with documented errors, which are necessary to accurately model the neutronics and thermodynamics of the blanket. This work aims to start filling this knowledge gap by measuring the density of ClLiF (LiCl + LiF) and FLiPb (LiF + Pb_2F) molten salts through the hydrostatic method. The weight of a nickel bobber was measured while submerged in the molten salt and compared with its weight in the air, and the differences in these weights was used to determine the density of the salt. This method was calibrated with high density organic liquids of precisely-known densities.. The densities were measured for various compositions of each salt from 20 °C over each salt composition's respective melting temperature to 800 °C. The errors from salt temperature, bobber weight, and bobber volume were documented and propagated through the density calculation. A polynomial was fitted to the data to form a density correlation based on salt temperature and composition. A comparison between results from this work and others from the literature was made. Additionally, a simple toroidal tritium-breeding blanket was modeled with OpenMC for various compositions of the various salts with their measured densities, and the tritium breeding ratio (TBR) and energy multiplication of each case were evaluated and compared to models using previous density estimates for each salt. Lastly, the effect of the uncertainty of the density measurement on the TBR and energy multiplication was quantified.

Dynamic simulation of different cryoplant options for the Divertor Tokamak Test (DTT)

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The "European Research Roadmap to the Realization of Fusion Energy" highlights the heat exhaust as one of the leading challenges in the development of fusion power plants. To address this issue, a dedicated facility, i.e. the Divertor Tokamak Test (DTT) facility, is currently at the early stages of its construction at the ENEA Frascati Research Center in Italy.
To replicate plasma scenarios relevant to DEMO operation, the DTT tokamak will rely on a superconducting magnet system featuring low critical temperature, cable-in-conduit superconductors. This system will include 18 toroidal-field coils (providing a magnetic field of 5.85 T), 6 poloidal-field coils (for the shaping and stabilization of the plasma), and 6 central solenoid modules, all of which will be actively cooled by means of supercritical helium (SHe) at approximately 4.3 K. Furthermore, the superconducting feeders, cryopump panels, and thermal anchors of the gravity supports of the magnet system will also be cooled by SHe at 4.3 K. Meanwhile, the thermal shields and high-temperature superconductors associated with the current leads will operate at temperatures of 80 K and 50 K, respectively. A large-scale cryogenic plant (or cryoplant), capable of guaranteeing an estimated cooling power of 11 kW equivalent at 4.5 K, will be necessary to meet the cooling demands of these systems.
This work employs newly developed, dynamic, system-level models to support the analysis of the forthcoming proposals for the DTT He refrigerator configuration. The target is to minimize the cooling power requirements and, consequently, the operational costs of the system, still keeping the necessary flexibility envisaged in an experimental facility. The study adopts models of standard cryogenic plant components—such as cold compressors, heat exchangers, and thermal buffers—available in the CryoModelica library of the 4C code. These models are employed to perform simulations of the DTT cryoplant performance for different transients and layouts, and to scan different values of its operational parameters to optimize the design. By comparing these variations, the resulting system-level analysis provides useful insights to support the choice of the optimal cryoplant design for the DTT facility.

Interface management in the development of the DTT RH Facility

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Due to the high complexity of Remote Handling (RH) procedures in fusion reactors, a virtual-only Verification & Validation process is insufficient to ensure correct operations of RH equipment. Physical tests in dedicated facilities are essential for validating and commissioning the robots, as well as training human operators. The DTT RH Facility is currently under procurement as part of the Divertor Tokamak Test (DTT) project. Its requirements and functions have been defined, and a preliminary logical architecture has been designed [1]. The Facility is divided into four main sub-systems: Buildings & Services, Mockups, Robots, and Control System. Given the complexity of integrating multiple subsystems, ensuring successful interface integration has been a key focus throughout the tender processes.

Following the previously presented conceptual design [2], this paper discusses advancements in the final design of the Mockups sub-system and interface management between sub-systems. The Mockups accurately reproduce the real DTT components, enabling reliable testing of RH equipment and procedures. By leveraging a model-based approach, interface management and integration across all sub-systems, particularly between Robots and Control System, have been performed. The use of SysML (Systems Modeling Language) diagrams has proven effective in defining the final logical architecture to meet all stakeholders' requirements, leading to more efficient conflict resolution.

The final designs for the RH Facility's Robots and Control System, along with the manufacturing phases, are expected to be completed by 2025.

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Progress and improvement of negative triangularity diverted plasma shape control on the DIII-D Tokamak

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The DIII-D negative triangularity divertor (NTD) shape development campaign in 2023 faced numerous challenges due to the DIII-D poloidal field (PF) coil system as well as the operational space required to avoid vertical and MHD instabilities. Detailed investigation of the NTD shape control challenges revealed the insufficient power supplies for each PF coils as the primary cause. This resulted in the usage of different power supply and PF coil pairings for different shapes, also termed, as patch panels on DIII-D. The patch panels used during the 2023 campaign were determined to have two major issues: (1) the power supply used on the divertor PF coils were insufficient and did not allow as high of a current as desired and (2) no hardware constraint on the boundary flux was used, so the PF coil currents for the same shape varied in different discharges.

Prior to the NTD shape development campaign in 2024, General Atomics Tokamak System Toolbox (GA-TokSys) was extensively used off-line for the identification and inclusion of optimized algorithmic and hardware configurations for the NTD shapes. The primary focus for the application was the expansion of the achievable NTD plasma parameter operating space to the fullest extent possible given the existing DIII-D PF coil set and their respective actuating power supplies. A modified plasma breakdown strategy was proposed with a non-standard PF coil pair, permitting connection of additional power supplies to the coils closer to the vicinity of the X-point. The modified breakdown algorithm was implemented in the DIII-D plasma control system (PCS) and off-line closed-loop simulations were performed to confirm an achievable breakdown null using the non-standard PF coil pairs. The achieved breakdown null quality was only modestly reduced compared to the standard breakdown strategy. In addition to the modification of the plasma breakdown, a set of new hardware configurations were also proposed for constraining the total poloidal field coils contribution to the plasma boundary flux and thereby reducing the demand in PF coil currents. The selection of the hardware configuration and the tuning of the NTD plasma shape controller were evaluated off-line utilizing the closed-loop simulation between the DIII-D PCS and GA-TokSys non-linear plasma evolution code, GSevolve. After successful off-line verification and validation of the proposed approaches with GA-TokSys simulation library, dedicated experiments were performed for evaluating the performance of the modified plasma breakdown algorithm and exploration of wider range of NTD shapes and the range of plasma currents for a given NTD shape. Experiments confirmed the successful implementation of the new breakdown methodology. The selected hardware configuration was able to achieve a new NTD shape was developed with average negative triangularity of 0.3 with up to 1.2 MA of plasma current and 1 MW of injected power. In this shape at 0.8 MA with 2 MW of injected power, scans were conducted over a wide range of X-point height and divertor leg length.

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DT Gas System Design for Diamond Anvil Cell Muon-Catalyzed Fusion Experiment

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A tritium-deuterium mixture delivery system was designed and implemented for the Muon-catalyzed Fusion Experiment (MuFusE) at the Paul Scherrer Institute (PSI) in Switzerland. The MuFusE Collaboration aims to narrow the gap between theoretical predictions and experimental outcomes in deuterium-tritium (DT) muon-catalyzed fusion processes.
The primary process loop of the tritium handling system is crucial for the safe and efficient delivery of tritium to the muon-catalyzed fusion experiment. The components include depleted uranium beds for tritium storage with controlled release and minimal tritium inventory, gas input connections for helium and deuterium supply with precise pressure regulation, a cryostat with a cold finger for accurate temperature control and efficient gas transfer, a tritium compatible palladium silver permeator for maintaining high gas purity levels essential for fusion reactions, and an expansion tank for safety during cryogenic operations.
These components interact to serve the roles of tritium storage, mixture creation, purification, target loading, and gas recovery. The design and operation of the primary process loop ensure minimal tritium inventory, high purity levels, and operational flexibility, critical for achieving the experimental goals while maintaining stringent safety standards.

A modular approach for the design of a salt-to-salt heat exchanger with 3D-Printed Triply Periodic Minimal Surfaces for future fusion machines

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Liquid breeder concepts are under consideration in various magnetic-confinement fusion projects aiming for rapid commercialization, as using a coolant that also acts as a breeding material can maximize the Tritium Breeding Ratio by incorporating more breeding material into the first wall design. Among liquid breeders, molten salts such as Lithium-Beryllium Fluorides (FLiBe) are of particular interest. To enable coupling with a secondary circuit for power generation or hot fluid production, the molten salt must transfer heat through a heat exchanger (HX) that ideally minimizes tritium permeation.
This study explores the preliminary design of a salt-to-salt HX, focusing on the potential of Triply Periodic Minimal Surfaces (TPMS) as the interface between hot and cold fluids. TPMS are three-dimensional periodic surfaces generated using trigonometric functions. Their unique geometry, with no sharp edges and a high surface-to-volume ratio, significantly enhances heat transfer efficiency while maintaining low pressure drops. By slightly extruding these surfaces, TPMS can define separate, non-intersecting volumes, making them suitable for 3D-printed HX structures. Certain TPMS geometries, such as Gyroids or Diamonds, are especially effective for HX applications due to their peculiarity of dividing the space into equal, complementary volumes.
Considering the high mass flow rates of molten salts required on both sides of the HX, beside the use of parallel circuits an advanced modular design approach is proposed and adopted in this paper. This choice limits the dimension of the component, making TPMS-based designs feasible for 3D printing. For the FLiBe salt, given specified inlet and outlet temperatures and inlet conditions for the solar salt on the secondary side, a methodology is presented to scale from a single HX module to a modular system. The performance requirements for the single module was derived from the overall requested performance as a function of the number of modules adopted in the design, enabling the design of the entire system from the optimization of the single module. To check the capability of performing an accurate design of a single HX module, a Gyroid-based HX was designed using Computational Fluid Dynamics (CFD) simulations. The module was then fabricated via 3D printing and tested in a water-to-water experimental setup at Politecnico di Milano. The test results are presented and compared to the numerical predicition, showing the extent to which the simulations reliably capture the HX performance.

Neural Network-Enhanced COTSIM Predictive Capabilities for Fast DIII-D Simulations

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Sustaining fusion reactions in tokamaks requires heating plasma to thermonuclear temperatures while maintaining confinement and stability. Neutral beam injection (NBI) and radio frequency waves, such as electron cyclotron range of frequency (ECRF), are used for heating, current drive, or fueling, which in turn shape the plasma current, temperature, and density profiles. COTSIM, a control-oriented predictive code, has been enhanced with neural-network surrogate models for transport and sources. A neural network version of the updated Multi-Mode Model (MMM 9.0.10), known as MMMnet, predicts turbulent transport with high accuracy and significantly faster computation times compared to the traditional MMM [1]. Additionally, neoclassical transport contributions are calculated using the Chang-Hinton model. NubeamNet [2], a neural network version of the Monte Carlo NBI module, predicts the effects of NBI efficiently and with reduced computational overhead. A simplified Gaussian beam model has been implemented in COTSIM to compute electron cyclotron (EC) heating and current drive. This model incorporates the Gaussian beam structure and employs the cold plasma dispersion relation to calculate the beam trajectory while accounting for weakly relativistic effects. The EC power deposition, localized at resonance layers where the wave frequency matches the electron cyclotron frequency, depends on the location-varying magnetic field and is computed using Westerhof's model. Plasma resistivity is modeled using a simplified Spitzer approach, while the self-generated non-inductive bootstrap current, essential for steady-state tokamak operation, is calculated using the Sauter model. The equilibrium is solved using a fixed-boundary equilibrium solver. The height and width of the H-mode edge pedestal are determined using the PEDESTAL module, which incorporates an empirical model for the L-H transition based on a threshold for the power crossing the separatrix. This integrated modeling approach, together with neural-network-based surrogate models, significantly enhances COTSIM's predictive capabilities, enabling fast and accurate prediction of temperature and safety-factor profiles in DIII-D discharges. Comparisons with experimental data from DIII-D demonstrate the accuracy and reliability of the integrated model. These modeling developments are critical for advancing tokamak plasma research and optimizing fusion reactor design and operation.

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Investigation of corrosion behavior of crushed samples of chromium beryllide Be12Cr under conditions of isothermal heating up to 1000°C in the presence of water vapor in the purge gas

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Beryllium intermetallic compounds are considered as promising materials for use in various fields of science and energy due to their unique physical and chemical properties. The Be12Ti is a prime candidate for application as a neutron multiplier in projects utilizing the HCPB concept of solid-state breeder blankets, selected for DEMO and testing at ITER. This is due to its high melting point, resistance to radiation swelling, low activation and excellent corrosion resistance. However, in order to expand the range of applied materials, it becomes relevant to study alternative variants. One of such promising materials is chromium beryllide Be12Cr, which has similar characteristics to Be12Ti, such as high thermal and radiation resistance. Nevertheless, this material requires additional research, especially in terms of studying its corrosion resistance in a vapor-gas environment.
Crushed samples of chromium beryllide Be12Cr produced by JSC “Ulba Metallurgical Plant” (Ust-Kamenogorsk, Kazakhstan) were chosen as the object of study. In this work high-temperature corrosion of chromium beryllide Be12Cr during isothermal heating to 1000°C in the presence of water vapor in the purge gas was studied. The crushed beryllide samples were a heterogeneous fraction with particle size from 1 to 4 mm and different particle shapes. The use of crushed samples was motivated by the need to increase the area of contact between the material and the vapor-gas medium, which allows a more detailed study of corrosion processes at the microlevel.
High-temperature corrosion experiments with crushed Be12Cr beryllide samples were carried out on the TIGrA experimental complex, which allows thermogravimetric analysis (TG), differential scanning calorimetry (DSC), and mass spectrometric (MS) measurements in one experiment. In the course of the work, the dependences of the pressure change of corrosion products (H2, HD, D2) and the rate of change of mass of crushed Be12Cr samples at different temperature modes were determined.The temperature dependences of the effective constant of water vapor interaction with Be12Cr samples obtained during corrosion experiments have been established. In addition, post-corrosion material science studies were carried out with Be12Cr samples subjected to corrosion interaction with water vapor. These studies included the analysis of microstructural changes and X-ray diffraction analysis, allowing to evaluate the changes in the phase composition and structure of the material after exposure to the vapor-gas environment.
The research was funded by the Science Committee of the Ministry of Science and Higher Education of the Republic of Kazakhstan with Program number BR21882237.

Enhancing Nuclear Fusion Systems: How Model-Based Systems Engineering Solves Complex Design Challenges

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The complexity of next-generation fusion systems is becoming increasingly unmanageable, due to many interrelated subsystems, technologies and conflicting requirements. Design information lacks the overview, consistency and traceability that is needed to make effective decisions and prevent rework. This is a drawback of traditional engineering approaches that spread out design information over domain-specific documents, such as interface reports, 2D diagrams or requirements lists. There are often no mechanisms in place to manage interrelations between documents effectively.

Model-Based Systems Engineering (MBSE) is the key to managing this complexity. MBSE builds a single network model of the engineered system that contains all relevant design information. The model can be inspected through automatically generated views that are tailored to the needs of all people involved in the project. Because there exists no duplicate information, the model is always up-to-date and self-consistent. This cannot be said for traditional document-based approaches. Despite the potential benefits for design quality, communication efficiency and project performance, MBSE is not widely adopted in nuclear fusion engineering.

In this contribution, we present a novel MBSE method and tool for nuclear fusion subsystems. We first specify the engineered system's components, interfaces, functions and requirements in the Elephant Specification Language (ESL), a textual systems modelling language. Then, an ESL compiler converts the specification to a large-scale integrated data model. This process includes various consistency checks and verification steps. Finally, the model is visualized in cabling and piping diagrams, functional flow diagrams, requirements lists, interface reports and Dependency Structure Matrices (DSM). All of these views can be adapted in scope and depth to display the most relevant information. Any design change, applied to the ESL specification, leads to updated views within only several minutes.

We report how this approach has supported the development of the Visible Spectroscopy Reference System (VSRS), ITER's visible Bremsstrahlung diagnostic. As the VSRS matured over a period of four years, the model was continuously adapted to reflect the actual design state. The views supported communication with both internal and external stakeholders for port integration, design for modularity, interface management, cabling management, instrumentation and control (I&C) architecture and signal chains.

In a second demonstration, we show how model elements can be reused across multiple systems. ITER's In-Vessel Lighting System consists of nine subsystems that are spread over various ports. Each subsystem has an independent model, but uses components from a common technological basis. Leveraging such commonalities across subsystems has significant cost- and effort-saving potential.

The MBSE method has demonstrated particular value for systems engineers, project managers and external stakeholders. We recommend a wider adoption to nuclear fusion projects, where the method can be improved and tailored to specific needs. A step towards product family engineering is an important recommendation for application on the level of full fusion power plants.

Modeling and Simulation of the Columbia Tritium Extraction Experiment (CTEX) for Molten-Salt Liquid Breeding Blankets

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Monday Posters 3, Lobdell (Building W20 Room 208), June 23, 2025, 4:00 PM - 5:30 PM

A computational tool is being developed at Columbia University to investigate the feasibility of molten salts in a vacuum-sieve-tray (VST) tritium extraction system and evaluate its scalability to larger experiments and pilot plants. This tool supports the design of the CTEX testbed and predicts the extraction efficiencies of hydrogenated molten salts, with a focus on FLiNaK.

The simulation framework is divided into two phases: the mixing phase, where hydrogen dissolves into molten FLiNaK under controlled pressure and solubility constraints, and the transient extraction phase, where hydrogen separation dynamics are analyzed. Key factors such as hydrogen solubility, mixing efficiency, liquid flow rates, droplet formation, hydrogen diffusion, and chamber geometries are modeled to assess system performance. Further enhancements will incorporate the effects of oscillating droplet geometries on mass transfer, as well as coupled heat transfer mechanisms to improve mixing efficiency.

This approach provides insights into optimizing hydrogen recovery and system design for tritium breeding applications. Experimental data from CTEX will be used to validate and refine the tool, enabling accurate predictions of extraction efficiency for various VST geometries, hydrogenic gas inputs, and alternative liquid breeding materials. The integration of simulation and experimental data aims to advance multi-scale modeling and accelerate the development of VST systems.

This poster will highlight the governing physics of mixing and extraction dynamics, key assumptions, the CTEX testbed's role in the simulation, and initial results on mixing and extraction efficiencies.

DEMO divertor target mock-ups with K-doped tungsten laminates monoblocks

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Tungsten (W) is the most eligible Plasma Facing Material (PFM) for the European demonstration fusion power plant (EU-DEMO) divertor target. At plasma operative conditions, such as high temperatures, High Heat Flux (HHF) cycles and neutron irradiation, W is subject to recrystallization which causes embrittlement. A proper doping of W with Potassium (K) can improve the PFM properties. Nano K-bubbles dispersed at the W grain boundaries can increase irradiation resistance and can hinder the motion of grains, leading to suppression of recrystallization.
New K-doped W Laminates (KdWL) materials were developed by Field Assisted Sintering Technique (FAST). Water-cooled ITER-like divertor target mock-ups were fabricated using KdWL monoblocks. The KdWL monoblokcs were joined to CuCrZr ITER Grade (CuCrZr-IG) cooling pipes by hot radial pressing (HRP) with an Oxygen Free Electronic Grade copper (OFE-Cu, Cu) interlayer manufactured by casting. Non-destructive examination by ultrasonic inspection were performed to assess the fabrication process and to later analyse the KdWL mock-ups behaviour under HHF tests. KdWL mock-ups have demonstrated to withstand up to 2000 HHF cycles at 20 MW/m2. KdWL is an excellent candidate as armour material for a future water-cooled DEMO divertor target.

Parametrical modelling and assessment of tokamak maintainability

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The system design models of tokamak fusion reactors can be parameterized and coupled to enable holistic code-based exploration and optimization of tokamak geometry. System codes have been developed for the major systems of the tokamak, but some systems such as the Remote Maintenance (RM) system have not been extensively codified. Development of RM systems has largely remained a decoupled process. We explore parameterization of Computer Aided Design (CAD) models and feasibility assessment scripts as the means to introduce a method to assess maintainability as a quantifiable optimization goal in the early convergence loop of tokamak geometry. Several tokamak design points have been studied that provide reference cases to approximate and link input and output parameters for the RM system design. EUROfusion DEMO is used as a basis for generating the CAD model, but we remain agnostic to tokamak configuration variations while exploiting and linking historical design data. The aim of the presented work is to accelerate and harmonize the creation of variations of the tokamak geometry coupled with numerical maintainability evaluation for plant architecture assessments. Hierarchical skeleton-based modelling methodology is utilized to model a responsive parameterized tokamak including geometrical elements of the RM system and installation paths of In-Vessel Components. Parallel to the development of the CAD model we construct a script to collate and export RM feasibility assessment values. The RM feasibility assessment is based on a selection of individual feasibility criteria, and mapping of the criteria to geometrical elements of the CAD model. The criteria effects are normalized and combined to provide converged feasibility values for tokamak design points. Range of design points are generated with the parametrical model and assessed utilizing the assessment script.

Investigation of Odometry Correction for In-bore Visual Pipe Weld Seam Inspection

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Maintenance of DEMO breeding blanket includes the removal and replacement of plasma facing components. Due to the components having an active cooling loop, multiple coolant pipes need to be removed to allow access to the tokamak. The connection then needs to be reconnected using the replacement components. To fulfil the safety requirements, the welded connection needs to be inspected and approved for operation. Due to space restriction of DEMO Tokamak, welded connections need to be inspected from the inside of the pipe, which is called in-bore inspection. During the testing of in-bore inspection system, it is noted that the measurement result is affected by the quality of the odometry. Due to the complex layout of the pipe and slip between encoder wheel and pipe wall, odometry using only encoder delivers inconsistent result. Therefore, a correction method for the odometry is developed to circumvent this issue.
This study presents the result of measurement correction using IMU and visual odometry. The resulting measurement correction will then be compared with the original measurement data using only encoder. With the result from each visual inspection system, a comparative advantage and disadvantage analysis of each method. The result of the analysis is then used for further development and integration of the inspection system into the overall cut and weld concept

Reinforcement Learning-based Density Profile Control with Active Stability-limit Enforcement in Tokamaks

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Achieving high plasma density is crucial for maximizing fusion power output in tokamak devices like ITER and the Fusion Pilot Plant (FPP). However, increasing the density through methods like pellet injection and gas puffing raises the risk of exceeding stability limits, leading to the onset of magnetohydrodynamic (MHD) instabilities and disruptions. Therefore, enforcing stability constraints while regulating plasma density is critical. Conventional density-control approaches rely on using pre-designed density profiles that satisfy the density limits as references for feedback control. However, real-time changes in plasma operating conditions could shift the stability limits themselves, leading to the immediate breach of the stability limits and compromising safety. This work proposes a reinforcement learning-based controller that regulates the plasma density profile while actively enforcing a pre-selected stability limit. The controller prioritizes tracking a reference density profile when the stability limit is satisfied. However, whenever the stability limit shifts, it dynamically adjusts the density profile to avoid violations, sacrificing reference tracking if required. Two controller versions, enforcing the Greenwald limit and edge stability limit, were trained in a control-oriented one-dimensional environment. Feedback simulations demonstrate the controller's effectiveness in regulating the density profile and its robustness in satisfying the stability limit.

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Preliminary design of the disruption mitigation system integrating muti-injectors for high-parameter tokamak devices

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The next fusion device with a 5MA plasma current in China will be built and perform the first plasma discharge in 2027. For the plasma discharge with such parameters, once the disruption occurs, the local high thermal load, electromagnetic stress and runaway electrons will cause serious damage to the device. Therefore, a disruption mitigation system (DMS) integrating muti-injectors has been preliminary designed in order to mitigate the deleterious effect resulted from plasma disruption and to ensure the safe operation of tokamak device. The DMS consisted of three sets of shattered pellet injection system units with muti-injectors, totaling 22 injectors. One containing 10 injectors would be installed on middle port A, and the other would be installed on upper port G and K, each of which contains 6 injectors. As for every pellet injector, it can use cryogenic cooling provided by liquid helium to desublimate material gases in the gun barrel and form the cylindrical pellet. Every pellet injector included a pellet producer, a gas feeding system, a propellant gas suppressor, a pellet diagnostic system and 5 or 3 injectors shared a vacuum pumping system. And every injector can work independently to form the different material pellets (Ne, D2, H2, Ar, or the mixture of different gases) with D*L=20×30 mm, whose material can be adjusted according to the physics requirements. The formed pellet will be accelerated to the velocity of 500 m/s by using the high pressure He of >50 bar, and the propellant gas would be removed during the process of the pellet and the propellant gas passing through the suppressor. Finally, the pellet would travel down the transmission tube, and breaking into fragments by shatter at its end before injecting into the plasma. And the fragment plume and injected direction can be adjusted by the structure of shatter. Additionally, we also built a test stand and conducted related tests to validate key components of the disruption mitigation system.

Fueling modeling and control for ITER start of research operation

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ITER's Start of Research Operation (SRO) targets the generation of plasmas reaching ~7.5 MA for durations exceeding 100 seconds. Effective control of fueling, density, impurity dosing, edge-localized modes (ELMs), and H-mode transitions is critical. To support model-based controller design, the Gas Injection System (GIS) and Pellet Injection System (PIS) have been modeled so that a complete fueling control system can be developed and assessed to address ITER's complex requirements and challenges. These challenges involve lag-time due to lengthy gas lines, fueling efficiency decay at high plasma temperatures and densities, synchronization of multiple actuators, and balancing ELM pacing with fueling needs using an advanced Actuator Manager (AM).

The GIS and PIS models utilize 1-D particle transport models for the gas flow and diffusion through the pipe and the pellet transport through the Flight Tubes (FTs) which have been implemented within the Plasma Control System Simulation Platform (PCSSP). Furthermore, the particle deposition into the plasma as well as the plasma-neutral interactions are modeled through the RApid Plasma DENsity Simulator (RAPDENS). The results presented here demonstrate the feedforward and feedback control with the complete integration of the GIS, PIS, AM, RAPDENS, and support functions for modeling the smooth transitions between fueling modes, and effective handling of actuator failures. This paper presents the architectural design, simulation results, and future strategies for optimizing fueling control on ITER.

High-Efficiency RF Amplifiers with Variable Plasma Load Handling for Fusion Plasma Heating

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Nuclear fusion power reactors will require low-cost, high-efficiency, reliable, and easy-to-maintain power electronics for plasma heating. Current RF heating systems use vacuum tube linear amplifiers to send waves to plasma-heating antennas via transmission lines, where system efficiencies are below 50%. Time-varying plasma dynamics lead to impedance variations which can result in inefficient antenna-plasma coupling, causing reflections that can damage components along with additional waste heat. Switchmode RF amplifiers, which operate solid-state transistors as on-off switches with a network to create an RF output, have achieved very high efficiencies, with an ideal efficiency of 100% for some topologies. The ability to handle variations in load impedance by voltage and phase control with a suitable combining network could enable improved RF-plasma coupling over the whole plasma operation envelope. We are developing a High-efficiency Amplifier for RF Plasma (HARP) innovation that combines these two features into a system suited for plasma operation. Such a system is designed to improve wall-plug efficiencies and reliability of RF heating systems for nuclear fusion reactors. The HARP system consists of switchmode RF amplifiers (currently of the Class-E type) connected via a Reactance Steering Network (RSN) to handle plasma load impedances. For a specified voltage ratio and phase difference between the amplifiers in the two branches, the RSN splits the power flow in such a way through the branches that each RF amplifier sees a pure resistive impedance instead of the load impedance, which may be inductive or capacitive. Maintaining pure resistive load is critical to achieving high efficiency and reducing thermal stress for Class-E amplifiers. The RSN can enable higher efficiency operation over a range of load impedance values by tuning the voltage ratio and phase difference knobs, thus showing promise as a method to handle variable plasma loads. In our DOE SBIR Phase I, we demonstrated high-efficiency Class-E RF amplifier operation at 27.12 MHz, tested RSN operation with some representative loads from an inductively coupled plasma model we made, and simulated and tested power-combining of boards to scale up to higher power systems relevant to fusion applications. The next phase of our project is to develop the real-time control capability of the HARP system to handle plasma load temporal variations (with the goal of improving the overall coupling efficiency) and to scale up the RF power to make the system relevant for fusion plasma heating. The frequencies demonstrated for this innovation range from 100s of kHz to 10s of MHz, relevant to plasma heating techniques including ICRH and HHFW. High board efficiency and plasma coupling are critical to advancing fusion reactors by increasing their net energy efficiency.

Plasma control actuator management and virtual circuit development on MAST Upgrade

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The plasma control system for the MAST Upgrade experiment provides a flexible framework for multi-parameter plasma shape and gas input control via the concept of virtual actuators. In this scheme, multiple controllers can be employed in parallel, each of them driving a virtual actuator that is optimised for the parameter under control. The operator then has flexibility to vary the mapping of each virtual actuator to sharing coefficients for the physical plant actuators. This mapping capability exists in similar guises for both poloidal field (PF) circuits and gas injection valves but, due to its more varied parameter space, most scenario development has been focussed on virtual circuit design.
Many of the PF coils are positioned for flexible control of the divertor leg geometry, but their close (in-vessel) proximity to the plasma means that the change of coil currents needed to move the divertor leg are highly dependent on the location of the divertor leg with respect to the coils.
The first campaigns of MAST Upgrade have seen significant exploitation of the virtual circuit method to achieve a wide range of plasma shape scenarios and various techniques have been developed. The main workhorse has been based on local linearisation methods using the TokSys framework (in collaboration with General Atomics) for simulation and testing. These have proved successful in controlling principal shape and divertor parameters such as outer gap and outer strike point radius immediately upon deployment. An alternative design tool based on spherical harmonics has also been demonstrated to produce more generalised scenario-agnostic virtual circuits for use in certain conditions. This method has additionally been used to develop rebalancing circuits that can redistribute PF currents without disturbing the shape parameters to avoid or mitigate coil current saturation conditions. Work is now under way to develop real-time adaptive virtual circuits that track the plasma geometry and adjust the circuits accordingly, further reducing the operator workload and opening up the operating space. The next steps will move toward use of machine learning for automatic virtual circuit adaptation and the design of virtual circuits for additional and alternative shape parameter definitions, such as squareness and flux expansion.
Upgrades and improvements to the plasma control system are ongoing, including migration to a faster more capable hardware platform with greater capacity for additional control. The development of a Real Time Data Network will enable a wider range of diagnostic measurements to be transmitted to the plasma control system for incorporation into more sophisticated plasma state estimation, event detection and control schemes to support our international collaborations.
This work has been (part-) funded by the EPSRC Energy Programme [grant number EP/W006839/1].

CFD-based Prediction of Critical Heat Flux in Double Wall Tubes of WCLL-TBM

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Fusion power plants have the potential to be the solution to the energy crisis, but their design and safe operation pose significant technological challenges, one of them being heat removal at high temperatures under high heat flux conditions. This research focuses on predicting Critical Heat flux (CHF) using CFD in the Double wall tubes (DWTs) of the EU water-cooled lead lithium Test Blanket Module (WCLL-TBM) for ITER. Water is circulated through the First wall (FW) and DWTs to remove heat from TBM. These cooling elements are exposed to high heat fluxes up to 0.5 MW/m2. Critical heat flux (CHF) causes a sharp temperature rise, risking component failure. Accurate CHF prediction is vital for the safe operation of TBM. Generally, the prediction of CHF is made by using empirical correlations or look-up tables, but the accuracy of these predictions is often debatable as they are derived from historical data for straight pipes, and are highly dependent on operating conditions. On the other hand, advanced multiphase computational fluid dynamic CFD is capable of detailed analysis of flow boiling for a more extensive range of geometry and operating conditions. The Eulerian-Eulerian multiphase approach coupled with modified Rensselaer Polytechnic Institute (RPI) wall boiling models was adopted in the simulation of the boiling phenomena. Void fraction and temperature at the wall are the main indicators of CHF occurrence, therefore the accuracy of the CHF simulation relies on the selection of appropriate wall boiling models and the momentum closures. An assessment of various combinations of boiling and momentum closure models has been conducted, and their performance was systematically compared with available experimental data for straight vertical pipes. The most appropriate model identified from this optimization exercise was utilized to simulate the flow in the DWT. The flow boiling simulations indicate that the chance of occurrence of CHF is very low during normal operation, the maximum vapor fraction observed is only 3% at walls. Furthering the study, analysis to determine the CHF in the DWT was carried out. This exercise was conducted in two ways, increasing heat flux and decreasing mass flow rate. Simulation initiated with a low heat flux and then incrementally increased by 0.025MW/m2, allowing convergence at each step. This process was continued until a sudden rise in wall temperature was observed, indicating CHF occurrence. A similar approach was taken for mass flow rate by reducing the value until CHF was attained. The maximum vapor fraction of over 90% was detected (>90% CHF criteria) between the bend and outlet section. The simulation indicated a risk of CHF after the bend but only when heat flux is increased well beyond normal operating conditions or mass flows are reduced well below the nominal operating flow rate. Also, the CFD calculated CHF was verified against available empirical correlations. Therefore, the flow boiling analysis supports a semi-mechanistic approach to accurate prediction of two-phase flow, CHF, and maintaining safety margin within the WCLL-TBM. These CFD results provide confidence in the acceptability and accuracy of the thermal-hydraulic design of the WCLL-TBM.

On the effect of model uncertainty on SPARC operational scenario predictions

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This work shows how changing model uncertainty and fidelity impacts the location of the most promising operating points for tokamak experiments and power plants. SPARC is a tokamak designed to achieve burning plasma conditions [1]. Both empirically-based models used in conjunction with a Plasma Operating CONtour (POPCON) solver [1] and physics-based models [2] have been used its design and operation planning. First, Monte Carlo analysis is performed that draws from distributions for assumptions related to confinement, H-mode transitions, profile shape, and impurity concentrations. Then, multifidelity Bayesian optimization is leveraged to scan the importance of individual assumptions as well as small possible deviations from device design. Improving access to H-mode, reducing uncertainties on confinement, maintaining full planned auxiliary power availability, and reducing the core dilution as a result of edge-radiator impurities are found to be levers for obtaining planned or better performance. Expansion into the effect of medium-fidelity physics-based transport models instead of empirical scaling laws is ongoing.

[1] Creely A J, et al.. Overview of the SPARC tokamak. Journal of Plasma Physics 86, (2020).

[2] Rodriguez-Fernandez, P. et al. Predictions of core plasma performance for the SPARC tokamak. Journal of Plasma Physics 86, (2020).

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Forced Convection Loops for Fusion Energy Breeder Blanket Research at Idaho National Laboratory

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Tritium breeding is fundamentally required for a sustainable deuterium-tritium fusion fuel cycle. Breeder concepts are divided into solid and liquid media. In each case, tritium must be harvested from the breeding medium. Tritium extraction from solid breeders typically relies on a sweep gas, such as helium with water and hydrogen constituents, to carry away tritium from lithium-containing ceramic materials. Liquid breeders produce tritium from lithium-containing liquids such as the liquid metal lead-lithium (PbLi), molten salt (FLiBe), and liquid metal pure lithium (Li). Two experiments featuring liquid breeder materials in forced convection loops are operating at the Safety and Tritium Applied Research Facility at Idaho National Laboratory. These experiments are equipped to provide essential validation data for tritium transport in candidate liquid breeders for informing and advancing extraction concepts. The first experiment is the Tritium Extraction eXperiment (TEX), a liquid metal PbLi loop that is testing the vacuum permeator tritium extraction concept. Currently a 1000-mm long, 12.7-mm outer diameter, and 0.5 mm wall thickness vanadium tube is installed in the test section. Two helical permeation window sensors made of pure iron are installed upstream and downstream of the test section in diverging wye-branch sensor ports to measure extraction efficiency. A moving magnet pump and vortex flow meter are installed to pump and measure the PbLi flow rate. A plenum with a helium cover gas enables the measurement of evolution rates into the cover gas. A hydrogen injection system is installed with four closed-end pure iron tubes injecting deuterium into the PbLi through permeation. The second experiment is the Molten Salt Tritium Transport Experiment (MSTTE), a fluoride salt loop designed to measure tritium permeation through structural 316 stainless steel tubing in a flowing salt system. MSTTE is currently filled with molten salt FLiNaK. The molten salt pump is a canned-rotor pump supplied by Copenhagen Atomics. Deuterium is injected in a similar permeation method to TEX. This presentation highlights the capabilities, specifications, status, and preliminary data of these two forced-convection loop experiments.

Mapping measurable parameters to tritium movement in breeding blankets using the MOOSE software

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A tritium breeding blanket is an essential technology that internally covers fusion devices to produce elevated amounts of tritium during operation. For commercial fusion energy, the blanket system must maintain a Tritium Breeding Ratio (TBR) appreciably greater than 1 for fuel cycle sustainability. The blanket must also withstand the extreme device environment and allow for tritium extraction. Additionally, in an accident scenario, understanding how and where tritium is released is necessary to quantify the maximum possible radiological release. Because of these challenges, it is important to accurately model tritium transportation within the fusion device to motivate improved functionality and safety in the blanket subsystem. Experimentally measuring tritium online is difficult and unreliable for long-term research capabilities. Because of this complication, an accurate pairing of measurable system parameters to tritium production and movement in the blanket channels using a simulated model is critical for the increased technology readiness level of fusion breeding blankets. Multiphysics Object-Oriented Simulation Environment (MOOSE) is an object-oriented multiphysics finite element software that can model complex phenomena in nuclear devices while consuming reduced computational resources. We are developing a model using MOOSE heat transfer and thermal-hydraulic solvers to determine how variation in device power history affects measurable channel flow, coolant temperature, and neutron/gamma flux for a given blanket geometry under different operational scenarios. Furthermore, we are evaluating how these changes can be linked to understanding the in-situ tritium inventory.

The blanket geometry used in this work models the coolant lead lithium (DCLL) blanket from the fusion nuclear science facility (FNSF) design. The DCLL blanket consists of sixteen channels with a PbLi breeder, helium coolant, and reduced-activation ferritic-martensitic (RAFM) steel structure. This model expands upon previous work at Oak Ridge National Laboratory. The present paper gives an overview of the demonstration blanket model and analysis methodology, along with preliminary results on the link between operational history and temperature/flow distributions.

Future work will link the measured parameters to the tritium movement within the blanket using the TMAP8 MOOSE application. The iterated model will also couple in neutronic modeling through Cardinal and OpenMC, which will allow simulation of the neutron/gamma flux detectors. The research completed is an initial step toward an accurate model for tritium tracking in a fusion system that will be used for inventory data, design improvement, non-proliferation, accident analysis, and breeding efficiency.

The FLARE High-Flux, Steady-State DT Neutron Source Facility

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The Fusion Linear Accelerator for Radiation Effects (FLARE) is a high flux, steady state, 14-MeV neutron source facility developed by SHINE Technologies. The FLARE facility provides the most intense steady-state DT fusion neutron flux available, making it a unique and powerful facility for R&D testing in support of fusion power technologies, including development and testing of radiation-hardened diagnostics and sensors, testing of tritium breeding blanket technologies, fusion product monitoring and other diagnostics, and DT fuel cycle studies.

FLARE is comprised of a neutron generator, a tritium purification system (TPS), an irradiation bunker, and other essential infrastructure. The neutron generator creates a steady-state deuterium (D+) ion beam which is then accelerated to energies as high as 315 keV and magnetically focused into a target gas chamber filled with gaseous tritium (T_2). A fraction of these ions undergo a DT fusion reaction, producing high-energy (14-MeV) neutrons.

As the D+ beam slows down through the gas target due to electron interactions, it results in a line source of neutrons of varying intensity over its length of ~100 cm, with a $1/r$^1.3 radial dependence in neutron flux. This geometry produces a large volume with high spatial uniformity in neutron flux available for testing. FLARE routinely operates with an accessible DT neutron flux of ~ 5×10^9 n/cm²s (over 10^{14} n/cm² over one operating day) and fast neutron flux levels of up to 1.8×10^{10} n/cm²s have been measured with this technology at a maximum DT neutron output of 4.6×10^{13} n/s.

The TPS continuously recovers mixed deuterium and tritium gas, cleans and isotopically separates the tritium, and returns a purified steam of T_2 to maintain target gas pressure in the neutron source, allowing for long-duration, steady-state operation. A differential pumping system is used to maintain the pressure differential between the gaseous target and the accelerator region which must remain approximately one-million-fold lower pressure to operate in a stable manner.

SHINE's DT fusion neutron source technology will be deployed as the driver of a testbed facility for tritium breeding experiments at the UK Atomic Energy Authority's Culham Campus as part of the Lithium Tritium Breeding Innovation (LIBRTI) program.

First measurement of the velocity slip coefficient and the tangential momentum accommodation coefficient (TMAC) of pure tritium

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The tangential momentum accommodation coefficient (TMAC) describes to what degree the reflection of a molecule in a surface interaction is specular or diffuse, and is therefore an important quantity for gas dynamics simulations and tritium processing in future fusion power plants. To the best knowledge of the authors, up to now, no TMAC values for tritium can be found in literature: neither ab initio calculated, nor experimentally obtained. At the Tritium Laboratory Karlsruhe, with a license to handle up to 40g of tritium, we have set up the CryoViMA experiment aimed at measuring the viscosity of tritium. The CryoViMA setup is based on a spinning rotor gauge, with a stainless steel rotor. The measurement principle and accuracy has been verified using helium, hydrogen and deuterium. In this measurement, the deceleration rate (DCR) of the spinning rotor is measured in dependence of the pressure of the surrounding dilute gas. In addition to the viscosity, the analysis of the data acquired in this manner also allows access to the velocity slip coefficient, from which the TMAC can be extracted as described in [1]. The CryoViMA setup allows for cooling over a wide temperature range using evaporated liquid nitrogen. With slow thermal cycling, this system enables a high-resolution measurement of the temperature dependence of the TMAC between 110 K and 300 K. In this contribution we will present first experimental values for the velocity slip coefficient and the TMAC of tritium on stainless steel in a temperature range from 110 K to 300 K.

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First Results of Corrugated Circular Waveguide Fabrication by Copper Cold Spray Additive Manufacturing

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Corrugated circular waveguides are key radio frequency (RF) components in electron cyclotron plasma heating systems for fusion devices, such as tokamaks and stellarators, as they allow low loss transmission of millimeter waves over long distances (>10m) between RF sources and in-vessel launching structures. Cold spray (CS) additive manufacturing is a process in which micro-particles are accelerated by a gas jet to impinge onto a substrate to form a coating, typically with high density and low oxidation. Due to the high deposition rate, relative low cost, excellent substrate/deposition interface quality, and geometric scalability, CS is a promising technology for the manufacturing of long (>3m) sections of corrugated waveguide. For producing corrugated waveguides using this methodology, spraying copper onto an externally corrugated cylindrical substrate followed by a chemical removal of the substrate to produce an internally corrugated structure is proposed using a VRC Gen III high pressure CS system controlled by a Fanuc M-710iC 6-axis robotic arm. First results of copper depositions onto both flat and cylindrical corrugated substrates representative of 100-200 GHz waveguide are presented; material porosity, grain structure, surface roughness and RF losses are reported for various spray parameters, post processing, substrate material, and corrugation aspect ratios.

Improving resistance to helium ion irradiation of tungsten with second phase dispersoids

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Monday Posters 3, Lobdell (Building W20 Room 208), June 23, 2025, 4:00 PM - 5:30 PM

Tungsten (W) has been selected as the material for plasma facing components (PFCs) in various tokamak devices due to its excellent thermal conductivity, high sputtering resistance, and low tritium retention. Despite these advantages, W-based PFCs face notable challenges during long-term operation in fusion reactors, such as embrittlement during temperate cycling, high-dose neutron damage, and high-fluence helium (He) ions in tokamak divertors [1]. Incorporating second-phase carbide or oxide particles into the W matrix has been demonstrated effective in suppressing the grain growth at high temperatures and improving the material ductility at lower temperatures [2].

In this presentation, we will show that dispersion phases can further improve the material resistance to He ion irradiation [3]. Dispersion strengthened W (DSW) embedded with micrometer-sized carbide dispersoids (e.g., TiC, ZrC, or TaC) were fabricated using spark plasma sintering (SPS), and then irradiated by 200 keV He ions up to a fluence of 4.5×1017 ions/cm-2 at 850 °C. Dense bubble formation was found inside the W grains and some dispersoids, with large, dense bubbles concentrated along W grain boundaries. In contrast, nearly no bubble could be observed at the W-dispersoid interfaces, and bubble density near the interfaces is lower than that in the grain interiors, suggesting that these interfaces can inhibit bubble growth. By tuning the sintering pressure, time, and temperature, we achieved finer grain sizes for both W and the dispersoids, reducing them to the scale of hundreds of nanometers. This refinement resulted in a similar suppression of bubble formation near the W-dispersoid interfaces. Thus, optimizing SPS parameters to refine grain size offers a promising strategy for improving the alloy's resistance to He irradiation. Meanwhile, significant elemental intermixing was observed near the W-dispersoid interfaces in some DSW alloys, such as those containing TiC. In these intermixed regions, helium-induced bubbles were evident, likely due to reduced melting points and increased vacancy mobility caused by intermixing. In addition, surface chemistry assessment revealed the enhanced protective role of ZrC, leaving metallic tungsten enriched from the surface to the subsurface region without any carbidization of tungsten, outperforming TiC and TaC. These findings emphasize the need for further optimization of the alloy composition and sintering process to enhance the performance of DSW.

This work is supported by US DOE under DE-SC00210005. Helium implantations were supported by the Center for Integrated Nanotechnologies (CINT), an Office of Science User Facility operated for the U.S. DOE Office of Science. Los Alamos National Laboratory, an affirmative action equal opportunity employer, is managed by Triad National Security, LLC for the U.S. Department of Energy's NNSA, under contract 89233218CNA000001.

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Development and characterization of Tritium Compatible Roughing Pump for Fusion Fuel Cycle

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Monday Posters 3, Lobdell (Building W20 Room 208), June 23, 2025, 4:00 PM - 5:30 PM

Kyoto Fusioneering in collaboration with Canadian Nuclear Laboratories, under the auspice of Fusion Fuel Cycles Inc. is developing the Exhaust system for Fusion Fuel Cycle.
Fusion reactors require large dry vacuum and transfer pumps for a variety of applications, including backing and roughing for torus evacuation, fuel cycle gas transfer and processing, and facility vacuum for safety systems. Commercially available mechanical oil-free pumps have been used successfully in tritium-bearing gas facilities. However, oil-free scroll pumps have been found to have low pumping speeds for hydrogen gases at high discharge pressures, which presents a challenge to the long-term operability and maintenance strategy of larger pumps when used with tritium.
To overcome these challenges, the Tritium Process Laboratory of the Japan Atomic Energy Research Institute, in collaboration with Mikuni Engineering, has developed mechanical oil-free reciprocating pumps for various tritium services. The main advantage of reciprocating-type pumps is their expected species-independent evacuation performance for all gas species over a wide range of discharge pressures. The developed pump is a prime candidate to be used with ITER vacuum roughing system for heavy tritium use. However, the original model suffered from defects such as wear of the piston and inner valve ring during prolonged operation, which led to a new model developed by Kyoto Fusioneering and Mikuni Engineering. This new model solves these problems and offers improved pumping capacity.
In this presentation, we report on the pumping exhaust characteristics of our newly developed reciprocating pump using light gases such as He and Q_2, which are commonly used in the fusion fuel cycle. Our findings support the installation of these pumps in future fusion plants.

Overview, Construction Status, and Engineering Challenges of the Vacuum Systems for the Material Plasma Exposure eXperiment (MPEX)

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Monday Posters 3, Lobdell (Building W20 Room 208), June 23, 2025, 4:00 PM - 5:30 PM

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 106 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 further diagnostic examination of the tested material. Due to the operating environment and performance requirements of MPEX, many engineering challenges came up during the design and fabrication 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, to develop and manufacture the vacuum systems for MPEX.

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.

Interpretable machine learning method for fast and accurate prediction of neoclassical toroidal viscosity torque in tokamaks

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Monday Posters 3, Lobdell (Building W20 Room 208), June 23, 2025, 4:00 PM - 5:30 PM

Neoclassical toroidal viscosity (NTV) torque plays a crucial role in influencing the toroidal rotation of tokamak plasma, thereby impacting plasma instability and overall performance. Accurate numerical simulation of NTV torque is essential for experimental design and understanding of the relevant physics processes. However, the high computational cost of calculating NTV torque and the black-box nature of end-to-end machine learning (ML) surrogate models present significant challenges in practical applications. In this study, an ML method for NTV torque computation was constructed by utilizing Bayesian parameter optimization strategy to explore the best ML model parameters, based on a comprehensive dataset obtained from extensive numerical simulations of NTVTOK and MARS-F codes under a variety of plasma conditions of the Experimental Advanced Superconducting Tokamak (EAST). The resulting optimal model, based on Extreme Gradient Boosting (XGBoost) algorithm, achieved high accuracy with coefficient of determination (R^2) of 0.99 in predicting NTV torque. Moreover, the inference time was reduced by three orders of magnitudes compared to the traditional numerical code. In addition, explainable artificial intelligence (XAI) technique is employed to provide physics insights for the model prediction and improve model transparency. This work offers an interpretable machine learning approach for fast and high-fidelity prediction of NTV torque, thus enabling the efficient modeling of plasma momentum transport in tokamak experiments.

Magnetic Field Compatibility Experiments of Ultrafast, Eddy Current Driven Actuators - ITER DMS Fast Shutter and ITER DMS Support Laboratory Propellant Valve Tests

Réfy D[1], Walcz E[1], Nagy D[1], Csiszár R[1], Zsákai A[1], Bartók G[1], Dibon M[2], La Bianca M[2], Unzhakov A[2], Bukhtiy D[2], Jachmich S[2], Kruezi U[2], Lopes C[2], Camuri M[2]
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Monday Posters 3, Lobdell (Building W20 Room 208), June 23, 2025, 4:00 PM - 5:30 PM

The ITER Disruption Mitigation System (DMS) [1] is a machine protection system for ITER, designed to mitigate plasma disruptions through the injection of cryogenic pellets [2]. This process involves high-pressure gas valves and ultrafast, eddy current driven actuators that must function reliably in the presence of strong magnetic fields near the tokamak. This study investigates the magnetic field compatibility of two key devices: the Propellant Valve and the ITER DMS Fast Shutter.
The Propellant Valve, developed by the HUN-REN Centre for Energy Research for the ITER DMS Support Laboratory Project [3], is a eddy current driven valve capable of releasing high pressure propellant gas within milliseconds, meeting the stringent requirements of laboratory Shattered Pellet Injector (SPI) systems [4][5]. The ITER DMS Fast Shutter, designed to mitigate premature disruptions caused by the propellant gas overtaking the pellets, employs an eddy current actuator to rapidly block the gas path [6][7][8]. Both devices were tested at the ITER Static Magnetic Field (SMF) test facility [9] to evaluate their performance under high magnetic field strengths, up to 275 mT.
This contribution will show the key performance parameters such as expelled gas consistency and wear behaviour of the Propellant Valve, as well as the closing time and wear of the shutter actuator under varying magnetic field conditions. The findings are essential for ensuring the reliability of ITER DMS components in a high magnetic field environment.

Keywords: ITER DMS, Eddy Driven Valve, Fast Shutter, Magnetic Field Testing, Static Magnetic Field Test Facility

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DIII-D HFS LHCD System Commissioning

Cengher M[1], Garcia I[1], Gould M[1], Leppink E[1], Lin Y[1], Pierson S[1], Ridzon J[1], Rutherford G[1], Seltzman A[1], Wukitch S[1]
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The DIII-D RF Systems is adding a new high field side (HFS) Lower Hybrid Current Drive (LHCD) system. The HFS LHCD system is expected to expand the range where the RF absorption into the plasma is possible beyond the limitations of the present ECH/ECCD system.

The DIII-D HFS LHCD system at 4.6 GHz will validate a novel approach for RF noninductive off-axis current drive for r/a~0.6-0.8. The centerpost placement of the launcher improves wave accessibility and penetration while reducing plasma interaction issues and associated coupler damage.

The 2 MW power system with eight 4.6 GHz klystrons has been installed. The high voltage power supply, cooling water circuits, and instrumentation and control commissioning has been completed, while the eight klystrons commissioning and conditioning is underway. All the eight klystrons were operated with power injection in the dummy loads with total power above 700 kW. The pulse length was extended to 3000 ms for five of the systems. Klystron and transmission line conditioning continues to increase pulse length and output power for the FY25 experiments.

The LHCD HFS launcher components and in-vessel waveguides were additively manufactured from GRCop-84. The waveguides and launcher assembly have been installed on the DIII-D centerpost, and the dedicated diagnostics that will be used to monitor the plasma conditions of interest, the launcher status, and the RF effect on plasma have been installed. The LHCD instrumentation and diagnostics include optical arc detection, impurity monitor for plasma-material interaction, pyrometer to monitor thermal response, thermocouples, visible camera monitors, Langmuir probes, and reflectometer for density profile measurements. The latest progress on the DIII-D HFS LHCD System commissioning will be presented.

Work supported by the U.S. DoE, Office of Science, Office of Fusion Energy Sciences, using User Facility DIII-D, under Award Number DE-FC02-04ER54698 and by US DoE Contract No. DE-SC0014264.

Engineering design of an automated facility at MIT for cryogenic neutron irradiation and testing of REBCO superconducting tapes for fusion devices

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Monday Posters 3, Lobdell (Building W20 Room 208), June 23, 2025, 4:00 PM - 5:30 PM

Large-scale superconducting magnets using Rare Earth Barium Copper Oxide (REBCO) high temperature superconductor (HTS) are the key to compact net-energy magnetic fusion devices. Such devices require REBCO magnets to operate within intense fast neutron and gamma radiation fields, presenting engineering challenges. Of principal concern is the exposure to lifetime neutron fluences in excess of 10^{18} cm^{-2} with kinetic energies (E_kin) greater than 0.1 MeV. The impact of these neutron fluence levels on REBCO critical current (Ic), critical temperature (Tc), n-value (n), and operational lifetime has not been well characterized at cryogenic temperatures (20 – 40 K) expected for superconducting fusion magnets. Cryogenic proton irradiation work has shown the need to follow cryogenic irradiation with in situ characterization of Tc and Ic at cryogenic temperatures with external magnetic fields without warm-up to room temperature, which can alter the microstructure and bulk superconducting properties through thermal annealing. A facility for REBCO cryogenic neutron irradiation and testing is being designed and will be constructed at the MIT 6 MW fission test reactor (MITR-II) and target irradiation of multiple REBCO tapes to total neutron fluences of $6x10^{18}$ cm^{-2}(E_kin > 0.1 MeV) in a 2-month campaign. Ic and Tc testing without warm-up is done by integrating instrumentation and current leads into a REBCO irradiation chamber with external magnetic fields provided by a 14 T magnet. Once the REBCO is installed in the chamber and cooled to the target temperature, the irradiation and testing procedure is as follows. First, the chamber is positioned so the REBCO is located within the bore of the magnet and the samples are characterized between 0 and 14 T at multiple magnetic field angles (θ) and temperatures (T). Then, while kept at cryogenic temperatures, the chamber is moved into position to expose the samples to fast neutrons for 1 week. Post-irradiation, the chamber is positioned with the samples within the magnet bore for characterization. This process is repeated over a two-month irradiation campaign, resulting in Ic(B,T,θ) and Tc(B,I,θ) data as a function of fast neutron fluence above 0.1 MeV. The design of the facility and the approach to challenges associated will be presented. This includes maintaining the cryogenic temperatures (20 – 100 K) of the sample for over 2 months through all phases of irradiation, enabling a robust approach to obtain Ic and Tc measurements at different magnetic field angles, integrating instrumentation and fixtures that are compatible with the harsh radiation environment, respecting size constraints for tooling to fit into both MITR-II and the 14 T magnet bore, and designing the chamber and processes to minimize material activation and radiation dose to personnel.

Physics-Guided Deep Learning Surrogate for Real-time Control of Vertical Stability in Tokamak Plasmas

Kumar A[1], Clauser C[1], Golfinopoulos T[1], Carpanese F[2], Garnier D[3], Sweeney R[4], Boyer D[4], Nelson A[5], Saperstein A[1], Granetz R[1], Rea C[1], Wai J[4]
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Controlling vertical stability dynamics in Q > 1 DT fusion tokamaks such as SPARC[1], ARC[2], ITER[3], and STEP[4] is critical for maintaining stability and mitigating disruptions. This study presents a physics-guided machine learning framework designed to enhance real-time stabilization of vertical displacement events (VDEs) while addressing engineering constraints such as power supply limits and actuator response times.

The framework integrates transformer-based machine learning (ML) surrogate models pre-trained on SPARC simulations, Alcator C-Mod disruption database, and fine-tuned synthetic datasets from equilibrium solvers. The model predicts key vertical instability metrics for SPARC, such as rigid and non-rigid vertical growth rates, inductive stability margins, and maximum controllable displacements, with computational efficiency suitable for real-time control. Reduced-order plasma models[6] (e.g. MEQ-RZIp, MEQ-FGE) are incorporated to enforce physical consistency and optimize power allocation for PF coils under thermal and electrical constraints.

To address the variability across machines (such as ARC and ITER), we integrate transfer learning techniques, including multi-domain preconditioning and fine-tuning with machine-specific data. Our initial results demonstrate a <5% error in predicting vertical stability metrics for ARC, with sub-millisecond inference times on GPU hardware. This level of performance is attributed to the incorporation of the Fourier feature embeddings within our surrogates during transfer learning, which efficiently captures non-local dependencies in the plasma responses; while residual connections mitigate vanishing gradient flow issues during training. These findings underscore a pathway toward scalable and integrated vertical stability control solutions for future fusion power plants, where proximity control must address broader challenges, such as steady-state operation and multi-objective optimization. Further validation in experimental and prototype systems is planned to bridge the gap between research and deployment in fusion power plant environments.

This work is partly funded by Commonwealth Fusion Systems.

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In-operando Evolution of Thermal Diffusivity of SiC during Ion Irradiation

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The study presented is part of a wider effort to evaluate NITE-SiC/SiC as a candidate structural material in the Liquid Sandwich Vacuum Vessel (LSVV) concept for ARC-class fusion energy devices. The LSVV design proposes the use of a non-conductive structural material shell made of Silicon Carbide composites like Nano Infiltration Transient Eutectic Phase Silicon Carbide Matrix reinforced with Silicon Carbide fibers (NITE-SiC/SiC) enclosing a conductive liquid (like molten lead). This assembly is contained within the liquid immersion blanket.

We will present the results of the first stage of an experiment aimed at understanding the evolution of thermal diffusivity of cubic-SiC (CVD) during exposure to a fusion relevant neutron spectrum. It is known that the fusion neutrons produce significant transmutation products, the effect of which on thermal diffusivity is currently unknown. Single ion irradiations were performed on the CLASS accelerator at the Massachusetts Institute of Technology at intermediate temperatures with Si4+ and He2+ ions to understand the temperature and dose rate effects on thermal diffusivity evolution in the context of ion irradiation. Thermal diffusivity has been measured in-operando using Transient Grating Spectroscopy. Each irradiation has been performed twice, once with a 3.4 um grating spacing and once with a 6.4 um grating spacing. These results are important as they are the first in-operando measurement of a key material property which will give unique insights into the evolution and saturation of diffusivity as a function of dose. Moreover, by irradiating with small and large grating spacings, the system will be tuned to allow in-operando measurements of thermal diffusivity during dual ion irradiation in an effort to understand the effects of gaseous and metallic transmutants. This represents the second stage of the project.

Commonwealth Fusion System's path to power on the grid

Mumgaard R[1], SPARC Team, CFS Team
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Tuesday Plenary - Bob Mumgaard, Kresge Main Theater (Building W16, upstairs), June 24, 2025, 9:00 AM - 10:00 AM

Commonwealth Fusion Systems (CFS) has been executing a path to fusion power plants since before its inception in 2018, with a goal to put fusion electrons on the grid in the early 2030s and to rapidly scale fusion power production thereafter. CFS is executing a milestone-based approach to fusion power plants based on the high field tokamak. It is currently the largest and most capitalized private fusion company and has systematically developed fusion-specific capabilities and recruited a diverse and capable team from fusion and adjacent industries. CFS has chosen a technical path that leverages the science demonstrated in prior tokamaks, and couples this with high field magnets based on REBCO superconductors and compact simplified designs. Working with MIT, CFS demonstrated a large bore 20T non-insulated fusion magnet in 2021. CFS demonstrated high field pulsed magnets in 2024. We have now put both technologies into serial production in a dedicated factory. Our current project, SPARC, is a DT burning plasma based on these magnets and is designed to close the remaining plasma physics gaps for a power plant and provide an integrated demonstration of our project execution, supply chain, and manufacturing capabilities. SPARC construction started in 2021 and is approximately 60% complete and is on track to operate in 2026 and reach DT Q>1 operations in 2027. In parallel to completing SPARC, CFS is preparing plans for the first power plant, ARC, using the expertise and designs formed during SPARC. As part of this program, CFS is developing materials, blanket, maintenance, and tritium systems in collaborations with others. The first ARC will be built outside Richmond VA in collaboration with Dominion, a major US utility, and provide 400MW to the PJM grid for paying customers.

In parallel to developing its power products, CFS is making significant efforts towards developing a stand-alone, diversified, and robust fusion industry that can deploy commercial power plants around the world. We believe that this presents a unifying goal for the community and we actively work to make this a reality by partnering with diverse organizations, advocating for fusion, expanding the stakeholders, and evolving the ecosystem. We are excited by the recent pace the fusion endeavor can and must move.

Design of a liquid Lithium Blanket for a Spherical Tokamak Fusion pilot Plant

Naish J[1], Abdallah M[1], Godhani V[1]
[1]Tokamak Energy

Tuesday Parallel 1a - Blankets and Tritium Breeding I, Kresge Main Theater (Building W16, upstairs), June 24, 2025, 10:30 AM - 12:00 PM

Tokamak Energy is designing a Spherical Tokamak Fusion Pilot Plant (FPP) for integrated test and validations of technologies, systems and processes required for commercial fusion energy deployment. The FPP, which is targeting start of operations by 2035, will consist of an operationally relevant fusion environment. By exploiting the inherent plasma physics benefits of the Spherical Tokamak, the FPP will demonstrate scalable net power in a fully integrated system. Tokamak Energy and its FPP design efforts are supported by the U.S. Department of Energy's Milestone-Based Fusion Development Program. As this will be a deuterium tritium (DT) machine it is essential that the tokamak has a breeder blanket to create its own tritium fuel.

The low aspect ratio of spherical tokamaks increases the solid angle of the outboard breeding blanket and decreases the solid angle of the inboard central column. This geometric effect enables a viable outboard-only breeding blanket solution particularly when optimized with the choice of liquid lithium as the breeder material. The choice of liquid lithium opens an additional benefit: the reduction, or even the elimination of a requirement for lithium-6 enrichment.

The design of the liquid lithium breeder blanket has been carried out using a combination of Neutronics, computational fluid dynamics (CFD), Magnetohydrodynamics (MHD), and mechanical simulations. The work presented shows the details of the design process including the neutronics calculations using Monte Carlo radiation transport codes [GEANT4 , MCNP] the CFD and mechanical calculations using ANSYS software suit and the MHD calculations using analytical solutions and Open foam calculations.

Readying Tritium Breeding Blankets for Fusion Machines: Validating Designs Through Emerging Experimental Facilities

Baus C[1]
[1]Kyoto Fusioneering

Tuesday Parallel 1a - Blankets and Tritium Breeding I, Kresge Main Theater (Building W16, upstairs), June 24, 2025, 10:30 AM - 12:00 PM

The most ambitious Fusion Pilot Plant (FPP) programmes aim to begin operation in the early 2030s. Yet, as of 2025, no tritium breeding blanket—or even a blanket module—has been tested for its three fundamental functions: heat extraction, tritium breeding, and neutron shielding. Breeding blankets remain one of the most critical yet least technologically mature fusion components.

While new designs continue to emerge, few progress beyond the (pre-)conceptual stage, and even fewer undergo experimental validation. In this talk, I highlight recent successes in testing breeding blankets and the key gaps that remain. Emerging integrated testing facilities will advance understanding of thermomechanics, thermohydraulics, tritium breeding and extraction, and electromagnetic effects on medium-sized blanket modules. However, material damage, maintenance schemes, and methods to manufacture full-scale blankets remain largely reliant on computational models, with significant uncertainties due to poorly understood or complex physics.

I also present Kyoto Fusioneering's latest efforts to close these gaps and, most importantly, outline the most feasible path to deploying breeding blankets in the coming years.

Neutron irradiation experiments to support fusion blanket development

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Tuesday Parallel 1a - Blankets and Tritium Breeding I, Kresge Main Theater (Building W16, upstairs), June 24, 2025, 10:30 AM - 12:00 PM

Fusion blankets are fundamental to the success of fusion energy systems, serving three primary functions: power extraction, tritium breeding, and shielding for magnets. In the absence of a dedicated nuclear-blanket component test facility (n-BCTF), relevant nuclear testing can begin immediately by utilizing existing fission capabilities to advance blanket technology. This paper presents a comprehensive set of current irradiation experiments leveraging available capabilities to address power extraction, tritium breeding, and shielding requirements for fusion blankets. These experiments include: 1) a thermal convection loop and the Advanced Test Reactor (ATR) to study power extraction and irradiation-induced corrosion, 2) solid breeder irradiation in the Neutron RADiography reactor (NRAD) to understand tritium release behavior, 3) irradiation of solid breeder materials in the High Flux Isotope Reactor (HFIR) to investigate mechanical integrity, 4) irradiation of Li-6 enriched breeder specimens for neutronic benchmarking, 5) shielding requirements for high-temperature superconductor magnets under neutron irradiation in the Massachusetts Institute of Technology Reactor (MITR), and 6) the design of an n-BCTF utilizing a fission neutron source. Collectively, these experiments address critical challenges that have been difficult to investigate without an n-BCTF.

A holistic approach for the integral design of breeding blankets after 15 years of experience in the development of the European TBM program for ITER

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Tuesday Parallel 1a - Blankets and Tritium Breeding I, Kresge Main Theater (Building W16, upstairs), June 24, 2025, 10:30 AM - 12:00 PM

Breeding blanket (BB) design is key for the feasibility of a future fusion power plant connected to the grid. Yet, current developments can only be considered as pre-conceptual in most cases, with very low TRL levels.
The main functions of a breeding blanket could be summarized in (i) tritium breeding, for a self-sufficient fuel cycle, and (ii) heat recovery, for a commercially viable energy conversion to electric power. However, there is a myriad of other requirements a BB design must also comply with in addition to fulfilling these two main functions. These pose challenges involving a number of disciplines which demand approaching the design with an integrated view from the very beginning. Since the design of any future fusion power plant will be highly conditioned by the design of its breeding blanket concept, the authors believe this integrated BB design approach should be one of the starting points for any fusion reactor design initiative. Despite the significant growth in the last years of private and public initiatives to design and build future fusion reactors, available experience worldwide in actual BB design and the almost endless list of associated challenges remains quite limited. In this context, exploiting the experience and lessons learnt from the most advanced breeding blanket prototypes currently under development, those of the ITER Test Blanket Module (TBM) program, seems particularly sensible.
The paper presents some general reflections based on lessons learnt from the development of the European TBM program for ITER over the last 15 years in connection with some preliminary engineering activities carried out in parallel for the definition of breeding blanket concepts for future fusion power plants.

Advanced Gyrotrons for Fusion Power Plants

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Tuesday Parallel 1b - Heating and Current Drive, Sala de Puerto Rico (Building W20 Room 202), June 24, 2025, 10:30 AM - 12:00 PM

Electron cyclotron heating (ECH) will be a critical requirement for tokamak-based fusion power plants. It will require gyrotrons that are lower cost and more efficient than currently available. The efficiency must exceed 60% with the output power in waveguide and will require advanced power supply systems. Calabazas Creek Research, Inc. (CCR) is developing advanced gyrotron systems meeting the requirements for fusion power plants.
The United Kingdom Atomic Energy Authority estimates that each plant will require 150 – 300, one MW gyrotrons [1]. With 150 gyrotrons as currently available, the cost of the ECH power will exceed $2M/day, based on a wholesale electricity cost of $0.30/kWh. Current gyrotrons operate at 50% efficiency with a single stage of collector depression. Simulations indicate the CCR's thee-stage design will achieve efficiencies exceeding 70%. Conventional gyrotrons also output the RF power in a Gaussian mode, and a Mirror Optical Unit (MOU) couples this into the transmission line. CCR's direct coupler outputs the gyrotron's RF power in a 97% pure, HE11, waveguide mode. Eliminating the MOU will reduce the gyrotron station cost by approximately $500,000 [2]. The direct coupler reduces RF losses in the gyrotron by 2-4% and eliminates MOU losses of 4-8%. In a power plant, each percentage efficiency increase will save tens of thousands of dollars per hour.
The three-stage collector and direct coupler are estimated to increase delivered RF power efficiency by 21%. This will reduce the operational cost of an ECH system with 150 gyrotrons by $394K per day, not including the savings from reduced cooling costs.
The three-stage collector is an extension of the two-stage collector CCR built in 2007. The additional stage increases the simulated efficiency from 63% to more than 70%. The reduced power densities allow fabrication using standard, oxygen-free copper, and no collector sweeping is required. Current gyrotrons use hardened copper and continually sweep the electron beam to prevent thermal failure.
A gyrotron with three collector stages will require three voltage supplies and three current supplies, in addition to the cathode voltage and heater supplies. This will require management of more than a thousand power supplies in a fusion power plant. CCR and N.C. State University initiated development of Machine Learning (ML) for gyrotron power supply control. The initial development focused on the collector voltage supplies and demonstrated that the appropriate voltage could be determined within 3% of the optimum value.
This presentation will present results to date and describe future development plans.

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Advances in RF Heating and Current Drive Systems on WEST

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Tuesday Parallel 1b - Heating and Current Drive, Sala de Puerto Rico (Building W20 Room 202), June 24, 2025, 10:30 AM - 12:00 PM

The WEST superconducting tokamak, equipped with an actively cooled divertor, focuses its studies on preparing future tokamak operations in an all-tungsten (W) environment like ITER [1]. A notable feature is that the plasma heating systems installed and under installation are only RF (Radio-Frequency) systems. This paper provides an overview of recent progress made in achieving pulse durations close to 1000 seconds by generating plasma current using the LHCD (Low Hybrid Current Drive) system, while maintaining the inductive loop voltage (V-loop) close to zero. In parallel with these experiments, two new RF heating systems are being developed at WEST: an Ion Cyclotron TWA (Travelling Wave Array) antenna operating in the 52-62 MHz frequency range and an ECRH (Electron Cyclotron Resonance Heating) system with a power capability of 3 MW/ 1000 s operating at the frequency of 105 GHz. The TWA antenna concept would allow to operate with low power density and low voltage increasing the reliability of the system compares with traditional ICRF antennas. The ECRH system aims at increasing the margin to reach H-Mode regimes, controlling W-impurities in the plasma core and mitigating MHD instabilities. The TWA antenna design progress are presented in this contribution and a particular attention is paid to the development and installation of the new ECRH system. The WEST ECRH system will operate with three gyrotrons manufactured by THALES company [2]. The first gyrotron was conditioned on the new FULGOR gyrotron test bench (FUsion Long Pulse Gyrotron LabORatory) in KIT. Efficiencies close to 35% and 45% in non-depressed and depressed collector operation respectively, were achieved with an RF power of 1.2 MW at 105 GHz. A second gyrotron will be conditioned in KIT during the first quarter of 2025. Both gyrotrons installed and commissioned at WEST would be ready to deliver 2 MW during the 2025 WEST experimental campaigns, with the antenna equipped with three water-cooled steering mirrors. The ECRH system is being commissioned by the WEST team in close collaboration with experts from the European ECRH network. Finally, an overview of the plasma scenarios planned for the use of ECRH in WEST will also be presented.

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High Field Side Lower Hybrid Current Drive Experiment in DIII-D

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Tuesday Parallel 1b - Heating and Current Drive, Sala de Puerto Rico (Building W20 Room 202), June 24, 2025, 10:30 AM - 12:00 PM

For an economical, steady state tokamak fusion power plant, efficient off-axis current drive is a critical enabling technology. High field side lower hybrid current drive (HFS LHCD) is a potential tool for efficient off axis current drive. From simulations of high qmin, DIII-D discharges, efficient off-axis current at r/a~0.6-0.8 with peak current density up to 0.4 MA/m2 and 0.14 MA/MW coupled is achievable with n||~2.7 at 4.6 GHz.[1] From the HFS, LH waves are expected to have improved accessibility and penetration[2] and single pass absorption.[3] In preparation for experiments, the HFS scrape off layer density profile has been characterized and found to have steeper profiles and less fluctuations than the low field side. This is expected to mitigate coupling and plasma material interaction challenges.[2] Furthermore, the HFS SOL density profile can be accurately predicted using global plasma quantities using machine learning. A compact launcher, located behind the center post protection tiles, is designed to distribute power poloidally utilizing a traveling wave, 4-way splitter and toroidally with a multi-junction that determines the launched wave spectrum.[4] To minimize reflections and electric fields in the coupler, each aperture has an integrated matching structure. The coupler is expected to couple 1.6 MW with a power density ~32 MW/m2. To simultaneously satisfy these machine requirements, a novel copper alloy, GRCop-84, developed for aerospace is used due to its favorable high strength, thermal and electrical conductivity, and readily manufactured using additive manufacturing (AM).[5] In the future, the Laves-phase precipitate, Nb2Cr4, can replaced by another suitable Laves-phase precipitate compatible with neutron environment, for example Ta2V4. Using AM, the embedded RF matching elements to ensure proper power splitting and minimize reflections, precise waveguide twists and bends can be readily manufactured. The AM primary limitations, component size and surface finish, were overcome through optimized joining techniques and incorporating chemical polishing. The klystrons have reached power and pulse length suitable to begin first plasma experiment. To minimize power conditioning, a vacuum window is located at the input of the poloidal splitter. The largest electric fields are in the phase shifters within the pressurized waveguide section. The vacuum window is a half wavelength alumina brick brazed into a CuCrZr window sleeve. To improve the brazing quality, the CuCrZr window sleeve is copper plated prior to the brazing to avoid issues associated with CrZr despite its low concentration. The latest simulations, design and system status will be presented. This will be the first application of LHCD from the HFS and provides an excellent opportunity to validate HFS RF wave physics, LHCD physics models, and RF AM technologies.

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Fusion power maturity levels and management process

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Tuesday Parallel 1c - Project Management, Systems Engineering, and Virtual Engineering, Kresge Little Theater (Building W16, downstairs), June 24, 2025, 10:30 AM - 12:00 PM

This work presents fusion power maturity level matrix and establishes standard definitions and best practices for using the matrix for a fusion powerplant design. The approach applies systems engineering thinking to the engineering design process.

When working with unknown and novel science, the development work and steps required for a design to reach concept level is not easily read across from other projects. Additional reviews/milestones are needed to ensure efficient and effective progression of the design, and due to the novel/new technologies being developed; there is no readily available precedent or industry best practice. Using guidance from the Royal Institute of British Architects as a well-known industry example in the built environment, there are only 2 levels before concept design is reached. This reflects the relative maturity of the construction industry and the use or re-use of 'standard' design solutions but is idealistic for a fusion plant. NASA Concept Maturity Levels framework was developed to measure space mission projects and becomes increasingly programme orientated and less appropriate for developing a fusion powerplant at higher maturity levels. Fusion Power Maturity levels (FP-ML) presented in this work are fusion relevant, and FP-ML will help to progress the design process in smaller chunks to help manage risks associated with new technologies.

Leveraging Systems Engineering in IFMIF-DONES

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Tuesday Parallel 1c - Project Management, Systems Engineering, and Virtual Engineering, Kresge Little Theater (Building W16, downstairs), June 24, 2025, 10:30 AM - 12:00 PM

Within the IFMIF-DONES (International Fusion Materials Irradiation Facility - Demo Oriented Neutron Source) project a classical systems engineering framework is tailored to the needs of neutron irradiation facility development [1]. DONES, an accelerator-based D-Li neutron source, aims to provide material testing under conditions simulating the first wall of future nuclear fusion reactors (such as EU DEMO). The complexity of the facility - highlighting main systems such as Test, Lithium and Accelerator Systems - necessitates a robust interdisciplinary approach to ensure integration and traceability across subsystems, researchers and involved industries alike.

The systems engineering framework employed for DONES merits not only from expertise in nuclear fusion but also from other projects like ESS (European Spallation Source). It implements Systems Engineering Body of Knowledge (SEBoK) principles [2]. A Plant Breakdown Structure (PBS) defines the system and subsystem boundaries while a Work Breakdown Structure (WBS) defines the main working fields. A dedicated Project Integration Team oversees this framework, facilitating traceability and consistency through detailed design documents, requirement matrices, and interface sheets. This approach ensures alignment with safety codes and standards, design objectives, and operational performance targets. A unique approach of establishing dedicated workforces (called Integration Work Groups) has been implemented for IFMIF-DONES to ensure the success of interdisciplinary fields, for example: accelerator alignment, diagnostics, operation and other groups.

Integrating digital tools, including CAD Configuration Models [3], Interface Management Systems and Requirements Management Systems enhances real-time collaboration across geographically dispersed research institutions and industry collaborators. These systems address challenges, such as inter-system dependencies. Furthermore, iterative design updates reflect ongoing research and adaptation to emerging requirements and interfaces. This systematic approach ensures the facility's readiness for commissioning and operational excellence while enabling its adaptation to future fusion research milestones.

In the scope of this work, I would like to present the Systems Engineering approach detailing the implementation of the SE foundations (structure and management rules) and requirements, interface, CAD configuration and interdisciplinary field management highlighting special cases unique to IFMIF-DONES.

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The first formal heat treatment of CRAFT TF high field coil

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Tuesday Parallel 1c - Project Management, Systems Engineering, and Virtual Engineering, Kresge Little Theater (Building W16, downstairs), June 24, 2025, 10:30 AM - 12:00 PM

Manufacturing a prototype toroidal field (TF) coil of the Comprehensive Research Facility for Fusion Technology (CRAFT) engineering is one of the key sub-projects of the China Fusion Engineering Test Reactor (CFETR) device. The CRAFT TF coil is designed with the capital letter "D", with maximum dimensions of approximately 19.5 m in length, 11.5 m in width, and 1.1 m in height. The high-field and medium-field windings of the TF coil consist of the Nb3Sn cable-in conduit conductor (CICC). Given that Nb3Sn superconductors are highly sensitive to stress and strain, the manufacture of TF coils will still adopt the "wind and react" process. Heat treatment is a crucial step in the manufacturing of TF Nb3Sn coils. To complete the heat treatment of the TF Nb3Sn coils, we designed the largest heat treatment system for superconducting coils in China, the full argon atmosphere oven heat treatment system. This paper will show the results of the first formal heat treatment experiment of CRAFT TF high field coil, including heat treatment system heat transfer fine model, coil deformation analysis, witness sample results, experimental temperature analysis, coil cleanliness analysis.

Risk and Accelerated Timelines for Fusion Projects

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Tuesday Parallel 1c - Project Management, Systems Engineering, and Virtual Engineering, Kresge Little Theater (Building W16, downstairs), June 24, 2025, 10:30 AM - 12:00 PM

In recent years, there has been a rise of private fusion endeavors developing projects to demonstrate their technologies. These projects are a crucial milestone in the commercialization process to prove a given concept, reduce commercialization risk, and attract further investment. Alongside the significant technical challenges associated with developing a fusion technology, start-ups face additional timeline pressures due to funding milestones and investor interests. While previous government-led fusion projects also faced timeline pressures, they often had flexibility to ensure project completion.

With limited peer examples, there are substantial unknowns regarding the risks of accelerated fusion projects. Hatch's experience with first-of-a-kind (FOAK) projects has shown that, despite caution, project delays, cost overruns, and cancellations can occur.

This paper presents learnings from Hatch's FOAK projects in the process engineering industry, nuclear projects, and fusion projects, as well as published case studies, focusing on the risk mitigation and, in some cases, the risk acceptance required for accelerated fusion projects.

Results from the first experimental campaigns of the “Junior” Levitated Dipole Experiment at OpenStar Technologies

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

OpenStar Technologies is a private fusion company exploring the levitated dipole concept for commercial fusion energy production. OpenStar has manufactured a new generation levitated dipole leveraging the advances made in high-temperature superconducting (HTS) magnet technologies since the last experiments of LDX. OpenStar’s first experiment called “Junior” aims to replicate and extend the results of LDX in a 5.2 m vacuum chamber with a modest ECRF power < 50 kW. Importantly, this experiment integrates novel HTS power supply technology on board the dipole magnet. Recently OpenStar has completed its first experimental campaigns with the Junior experiment, achieving first plasmas in late 2024. Experiments conducted with the full levitated system are planned for early 2025. This presentation provides an overview of the main results from these experiments and details improvements planned for future campaigns.

Machine learning-based anomaly detection for ITER's Tokamak Systems Monitor: a gyrotron case study

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

The Tokamak Systems Monitor (TSM) software will provide operators with timely assessments of machine health, component lifetime, and early warnings of potential faults. Among its key functions, the TSM will employ data-driven anomaly detection methods to identify unexpected or abnormal behavior across a broad range of systems and diagnostics. This work presents an initial proof-of-concept for anomaly detection applied to gyrotron pulses, leveraging data from the European gyrotron prototype. The method combines dimensionality reduction and clustering techniques to identify deviations from expected operational patterns. This approach enables the detection of subtle anomalies that might otherwise go unnoticed. While this first demonstration focuses on post-shot anomaly detection for gyrotrons, the methodology is designed to be adaptable to other systems and signals within the TSM, contributing to the improved maintenance strategies and reliability of the tokamak.

Enhancement of lithium isotope enrichment efficiencies in asymmetric multi-stage electrodialysis system

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

There will be a large demand of highly enriched 6Li compounds for breeding blanket of a fusion pilot plant. In the meanwhile, establishing Li enrichment process with no use of harmful element is a primary challenge to replace conventional chemical exchange process using mercury, which has environmental risks due to its toxicity. Electrodialysis can separate lithium isotopes by utilizing the mobility difference between ^{6}Li and ^{7}Li in a Li-ion conducting membrane. An efficient multistage design with less waste is needed to bridge a technological gap between today's experimental and the target engineering scale. However, detailed multi-stage design and optimum operational conditions remain still unclear. Here, numerical calculations for multi-stage systems are carried out to clarify key design parameters for an efficient enrichment system with reflux that can operate continuously.
The isotope separation factor in a single cell was experimentally investigated by applying voltage on a Li-conducting electrolyte that separates pure water and LiOH solution. As a pretreatment, a voltage of 2 V was applied for 24 hours at each experimental temperature to avoid possible influence of isotope bias existing in the membrane. After the electrodialysis tests, Li isotope ratios in the source (X_6,s and X_7,s) and condensed solutions (X_6,c and X_7,c) were quantified by inductively coupled plasma mass spectrometry (ICP-MS) to obtain heads separation factor ω. As a result of the single cell experiment, ▯ were obtained to be 1.056–1.073 in the temperature range of 293−323 K.
A calculation code for multi-stage electrolysis cells has been constructed to calculate concentration, isotope fraction, flow rate in each cell and the separation performances. Each electrolysis single cell was connected by two different ways, so called as cross- and straight- type. In symmetric multi-stage cells, the overall separation factor ω did not effectively increase with number of the stage (n); even with a very large system with n > 70, the overall separation factor in the cross and straight types were still below the experimental ▯ value in the single cell. Enhanced overall separation factor of cascade system, recovery ratio of ^{6}Li isotope, and required stage for 90% enrichment will be presented in the conference. These findings provide a new strategy to improve the overall efficiency and therefore provides design scheme for further scale up of the mercury-free enrichment method.

Competency Matrix for the ITER Ion Cyclotron and Electron Cyclotron Heat & Current Drive Transmission Line Systems

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

The fusion energy industry is evolving as technologies mature and demonstration facilities progress. As such, the need for an engineering base of expertise is rapidly growing. There is an opportunity to better define the competencies required to fulfill the demands of certain fields in fusion engineering. Specifically, this paper aims to characterize engineering competencies needed to effectively deliver the ITER Project contributions to Ion Cyclotron (IC) and Electron Cyclotron (EC) Heating & Current Drive (HCD) Transmission Line (TL) systems. In this application, the benefits include (1) a defined body of knowledge across functional roles, (2) a transparent development framework to attract and retain talent, and (3) a credible means to prepare the engineers for future opportunities in the fusion industry.

The paper will present a project-specific competency matrix that defines technical knowledge & skills that can be achieved through a tailored selection of education, key experiences, and training. The matrix will be both informed by past project experiences and forward-looking project needs. Multiple functional roles will be defined to support the range of work activities, including technology development, final design & manufacturing. A graded approach based on seniority and contribution expectations will be formulated to support the range of expected staff. Core technologies, key engineering principals and auxiliary systems will be considered.

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Impact of Poloidal NWL Non-Uniformity on Temperature Distribution in Water-Cooled Ceramic Breeder Blankets

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

A breeding blanket, a key subsystem of fusion tokamaks, converts fusion energy into thermal energy. Poloidal and toroidal variation in the solid angle between blanket modules and the plasma cause non-uniform neutron wall loading (NWL) across the modules. This NWL non-uniformity causes spatially non-uniform nuclear heat deposition along the poloidal direction, resulting in a gradient of volumetric heat generation within the blanket modules.
This study examines the impact of poloidal spatial resolution of heat source on the temperature distribution within a water-cooled ceramic breeder blanket module. For numerical analysis, a 3D box-shaped blanket module with a multilayer structure was modeled and analyzed using a magnetic coordinate system to analyze the non-uniform heat deposition in flux surfaces. The poloidal distribution of local heat generation was derived using a neutron-transport code (MCNP). Subsequently, we evaluated the effect of spatial resolution in local heat generation on temperature distribution using the ANSYS Fluent code. Among the four inboard blanket modules, we focused on the uppermost inboard blanket module, which is subject to the steepest poloidal NWL gradients. This is due to its position relative to the plasma and the significant variation in solid angle between the blanket surface and plasma. To analyze the impact of these gradients, three volumetric heat generation scenarios were considered: (1) uniform distribution, (2) 5-cell discretization along the poloidal direction, and (3) 40-cell discretization along the poloidal direction. In the uppermost blanket module, the tungsten layer, which is subject to the highest volumetric heat generation, exhibited a local heat generation difference of up to 11.6 MW/m³ between the uniform condition and 40-cell discretized condition. This highlights that employing a uniform heat generation can significant underestimate local heat sources in blanket modules. While poloidal variations in NWL significantly influence local temperature distributions, their impact on the average tungsten layer temperature remains minimal, with the uniform heat generation condition yielding 441.5°C compared to 442.1°C under the 40-cell discretized condition. Under the 40-cell discretized condition, local tungsten temperatures were up to 20.1°C higher than those predicted by the uniform assumption. These findings underscore the necessity of considering poloidal heat source variations in thermal design, especially to mitigate the risk of structural materials exceeding operational temperature limits. Overall, this study provides valuable guidelines for module-level thermal design of breeding blankets, enabling precise thermal behavior predictions and ensuring structural integrity under operational conditions.

Heat transfer and velocity distribution in liquid metal MHD flows around cooling pipes

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

The water-cooled lead lithium (WCLL) breeding blanket has been selected to be tested during the experimental campaign carried out in the International Thermonuclear Experimental Reactor, ITER. In this blanket concept, lead lithium (PbLi) eutectic alloy serves as tritium breeder, neutron multiplier and heat carrier. Volumetric heating in the liquid metal is produced by the high-energy fusion neutrons that release their kinetic energy via nuclear reactions with blanket materials. In the breeding zones of the WCLL blanket, which consist of regions delimited by first wall, back-plate and radial-toroidal stiffening plates, the thermal power is extracted by water-cooled pipes around which the PbLi circulates.
The features of the liquid metal flow in the blanket at fusion relevant parameters are determined mainly by the action of driving buoyancy forces, due to density gradients, and braking effects caused by viscosity and electromagnetic forces. The latter originate from the interaction of the moving electrically conducting PbLi with the plasma-confining magnetic field. The characteristics of the established magneto-convective flow, which affect heat and mass transfer in the blanket, depend on thermal conditions, such as differential or volumetric heating, geometry, electrical properties of the structural material, reciprocal orientation of gravity and magnetic field.
In the present study, we investigate numerically magneto-convective PbLi flow in closed geometries with volumetric thermal load or differential heating and internal cylindrical obstacles that represent cooling pipes. The effects of applied temperature gradient, strength and orientation of imposed magnetic field are studied via 3D numerical simulations. The aim of the analysis is getting an overview of flow features and heat transfer properties depending on operating conditions and geometrical set-ups. This could allow identifying the arrangements that are more favorable to heat transport in the breeding zones.

High temperature superconductor solder facility at MIT PSFC

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

Solder is used as a component in magnets containing rare-earth barium copper oxide (REBCO)-based high-temperature superconductors (HTS). It is an integral constituent that allows for efficient electrical, thermal, and mechanical connection between individual superconducting tapes. Solder vacuum-pressure-impregnated (VPI) HTS magnets are a critical technology in high-field fusion devices like the SPARC tokamak. Since 2020, the High-Temperature Superconductor Solder Facility at the Massachusetts Institute of Technology Plasma Science and Fusion Center (MIT PSFC) has been used to solder about 80 superconducting components, including the SPARC Toroidal Field Model Coil (TFMC) and Central Solenoid Model Coil (CSMC), as well as cold bus cables for the Superconducting Magnet Test Facility at the MIT PSFC, and a number of samples for process development. The facility builds upon prior process development at the center carried out in smaller satellite facilities in the years prior to the SPARC TFMC program. This broad experience has yielded a well-controlled, consistent, and reliable process. This poster will provide an overview of the facility's capabilities, as well as lessons learned from the multitude of magnets manufactured. Notable results will be described, including the successful soldering of the SPARC TFMC and CSMC, and research showing low performance degradation from exposure to SPARC-scale solder manufacturing processes.

Magnetic Permeability Study of 3D-Printed Austenitic Stainless Steel 316L Components

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

Austenitic stainless steel 316L, widely used in fusion experiment reactor vacuum vessel for its excellent corrosion resistance and mechanical properties, has garnered attention for additive manufacturing (AM) applications, particularly in critical industries such as aerospace, biomedical, and chemical processing. This study investigates the magnetic permeability of 3D-printed 316L components produced using selective laser melting (SLM) and other AM techniques. Magnetic permeability is a crucial parameter for ensuring material compatibility in applications requiring non-magnetic behavior, such as medical implants and electronic shielding.

The study evaluates the influence of print parameters, such as laser power, scanning speed, and post-processing treatments, on the microstructure and magnetic properties of 316L. Results reveal that 3D-printed 316L exhibits variations in magnetic permeability compared to its conventionally manufactured counterparts due to residual stresses, anisotropy, and phase transformations induced during the printing process. Post-processing techniques, including heat treatment, are analyzed for their effectiveness in reducing magnetic permeability and restoring the material’s non-magnetic nature.

This research provides critical insights into optimizing AM processes to meet stringent magnetic property requirements, paving the way for broader adoption of 3D-printed 316L components in sensitive applications. The findings also highlight the interplay between AM parameters and material properties, contributing to the development of standardized guidelines for 3D printing austenitic stainless steels.

Interpretation of Magnetic Diagnostics for Vertical Stability Control on NSTX-U

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

In tokamak reactors, vertical elongation of the plasma enables higher performance at the cost of requiring active stabilization of the plasma's vertical position. To achieve experimental flexibility in present-day tokamaks, the vertical stability control system must be able to recognize and respond to unstable vertical motion over a wide range of plasma conditions. This work is focused on optimizing the interpretation of the existing magnetic diagnostics on NSTX-U to improve the performance of the vertical stability controller and assess how the controller can be expected to perform in new scenarios such as those with negative triangularity shapes.

The NSTX-U vertical stability controller performs Proportional-Derivative control on the vertical position of the plasma as estimated from poloidal flux loop measurements. The NSTX-U database contains 43,170 well-converged EFIT01 equilibria from 803 plasma discharges during initial NSTX-U operation, which are used here as a reference for assessing the controller's performance. Comparing the plasma current centroid vertical position from these equilibria to the estimates of the vertical position made by the controller, the coefficient of determination is only -0.0671. This indicates that, on average, the vertical stability controller was not effective at determining the true vertical position of the plasma. In theory the controller may be able to stabilize the plasma position even without precisely knowing it, and therefore the time-series cross-correlation between the EFIT01 and controller estimates may serve as a better metric for comparison. However, the time-series analysis shows a positive correlation above 0.80 in only about 7% of discharges, leaving significant room for improvement in the interpretation of the magnetics diagnostics.

The performance of the vertical stability controller should improve if the model for estimating the vertical position can be improved. The existing model takes a weighted sum of poloidal flux measurements from 4 pairs of sensors to estimate the vertical position. By re-fitting the existing model to the EFIT01 data, the fit can be improved to have a coefficient of determination of 0.409 with a time-series cross-correlation above 0.80 for 25% of discharges. Further analysis shows that a near-perfect recreation of the plasma positions with a simple linear model requires knowledge of all coil and vessel currents, and is therefore not possible with the information available in real-time. To improve upon the existing model, a variety of new models are developed by considering different combinations of measurements from flux loops, Mirnov coils, and Rogowski coils, and adding increased mathematical complexity to the model such as nonlinear terms. Introducing nonlinear terms into the model leads to overfitting of the data. Using the linear model with all available magnetics signals produces a coefficient of determination of 0.85 and time-series cross-correlation above 0.80 for about 78% of discharge, a significant improvement over the previous model. Machine-learning is now being deployed to try to make further improvements to the vertical position model.

Measurement and assessment of Plasma-induced damage on Tungsten Divertor Tiles with the ITER In-Vessel Viewing System prototype (IVVS)

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The In-Vessel Viewing and Metrology System (IVVS) is a diagnostic tool for the ITER project, designed to inspect plasma-facing surfaces within the tokamak. Following the construction and validation of a full-scale prototype of the measurement probe compatible with the ITER radiation and vacuum environment in 2023, recent progress has showcased its application on a full-scale ITER divertor inner-vertical target (IVT) and on real, damaged tungsten Plasma Facing Unit (PFU) from CEA WEST that had been exposed to plasma and displayed deposition of films of tungsten on the surface. This deposition is currently being studied, as detachment of the films caused disruptions in the WEST tokamak, and so could have major implications for the ITER project or any tungsten-based tokamak.
The tests done by Bertin have successfully validated the performance of the IVVS at ITER- relevant viewing angles, distances, and focal parameters, over ITER-specific geometries, and at full cable/fiber length, confirming its suitability for in-vessel use. They have also shown the capabilities of the system to detect flaking of the tungsten with real tiles.
On the IVT measurement campaign, the system clearly detected geometrical features such as tile gaps and edges, with the metrology channel providing precise range measurements with precision on planar areas of 0.1 mm from 0.5-10m and angles from 0-60 degrees. Over the full IVT geometry, dimensional errors are <1 mm. Viewing measurements provided high-contrast images with spatial resolution limited by sampling density. The viewing channel was able to identify damage caused by heat treatment, even when the surface changes were sub-10 micron in dimension.
Additional measurements were conducted on other tungsten tiles that had been exposed to plasma, demonstrating the IVVS's capability to detect and analyze deposited films and other surface damages. These findings further validate the IVVS's suitability for monitoring the condition of the ITER first wall.

Real time exhaust/fuel mix monitoring for ITER and W7-X

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A real-time (RT) exhaust gas analyzer on ITER, also known as Diagnostic Residual Gas Analyzer (DRGA [1]) will sample the divertor pumping duct and thus, indirectly, the sub-divertor. Based on recent tokamak data, it is anticipated that the thus-measured, sub-divertor neutral gas composition will be representative of the corresponding ion species composition in the main plasma, over a few-seconds of plasma-wall equilibration times [2, 1]. This is a significant aspect of the physics basis of this diagnostic system. Toward its validation for ITER, time-dependent SOLPS-ITER simulations are performed in which the core boundary condition of 3He/(3He + D + T) concentration is abruptly changed, with the goal to deduce a measurable change in the divertor pumping gap on a timescale consistent with ~1-sec detection capability. The 3He concentration is a critical parameter for an ICRH scenario that is efficient in heating T ions in the core plasma [3]. Early simulations showed promising trends, albeit with the impurity neutrals assumed to remain quasi-steady state [4]. More accurate time-dependent simulations, capturing the dynamic response of both plasmas [5] and neutrals [6], will be presented here. In parallel, an emerging, collaborative effort on W7-X is seeking to use an early ITER DRGA prototype, now fully integrated into this largest stellarator in operation [7], to study how the non-axisymmetric, yet periodic, sub-divertor regions related and respond to core plasma compositional changes. The status of initial, synergistic (albeit still steady-state) modeling studies on W7-X, related to this RT exhaust analysis on W7-X, will also be summarized. This includes the validation of multi-scale, multi-species modeling using DIVGAS, adapted to W7-X [8], and recently extended to include helium (He) [9]. More recently, DIVGAS has been coupled successfully with EMC3-EIRENE for hydrogen plasmas [10]. Starting from the hydrogen exhaust analysis in ref. [11], also He exhaust is studied using EMC3-EIRENE and the coupling to DIVGAS for He exhaust is analyzed. An update on related experiments will also be included.

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Status Update of SPARC REMC Electrical Characteristics

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The SPARC Runaway Electron Mitigation Coil (REMC) is an asymmetric, in-vessel copper coil intended to prevent the build-up of relativistic “runaway” electrons during a plasma disruption. A sufficiently large population of runaway electrons can damage plasma-facing components and are a serious potential failure mode for future high-current, power-producing tokamaks. In SPARC, these high energy electrons could carry several megaamperes of current[1]. The REMC prevents the build up of runaway electrons by creating stochasticity and decreasing the electron confinement in the plasmas when current runs through the dominantly n=1 coil.

To avoid interfering with tokamak ramp-up and ramp-down, the SPARC REMC circuit includes a slow (~50 ms) switch. Recent ThinCurr analysis shows that the coil is expected to float to 1.75 kV relative to its grounded housing when the switch is open during a disruption. The REMC also contains several joints that will be exposed to a wide range of pressures due to the use of Massive Gas Injection during disruptions[2]. A robust electrical insulation strategy is required to prevent arcing either directly or through Paschen breakdown of the gas.

The REMC's uses several methods to prevent arcing to ground, including: spray coated alumina for the bulk of the coil, magnetic field as an insulator to inhibit arcing at the joints, and a resistor in parallel with the switch to limit the total in-vessel voltage build-up. This presentation will discuss the theoretical basis for improved resilience to voltage breakdown in the presence of a background magnetic field and present scalings for how Paschen curves are modified in the presence of a background magnetic field. Results of testing voltage breakdown at the REMC joints in the presence of a strong background magnetic field will also be discussed. Finally, the impact of a parallel resistor on REMC operation as well as other always-conducting REMC operational modes will also be discussed.

Work supported by Commonwealth Fusion Systems.

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Neutron Shielding of a Levitated Dipole Reactor Power Plant

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Levitated Dipole Reactors (LDRs) are an attractive option for magnetic confinement fusion power plants, offering desirable plasma physics and rapid iteration times when compared to other confinement schemes. An LDR consists of a magnetically levitated high temperature superconducting core magnet which is surrounded by the fusion plasma. A key challenge in proving the viability of the concept is showing that the system can be engineered to function in an environment with high D-T and D-D neutron flux. Three key requirements have been identified: 1) thermal inefficiencies require the total heating of the core magnet to be <1% of the thermal output of the reactor for the plant to achieve break even; 2) the total heating of the core magnet must be low enough to achieve commercially viable duty cycles; 3) the neutron degradation of the HTS must be slow enough to keep the cost of electricity competitive. Similar to other magnetic confinement concepts, adding neutron shielding decreases the performance of the reactor and therefore increases the overall device size and cost. Hence, the shape and material of the neutron shielding must be optimized to minimize this impact while still meeting the aforementioned key requirements. This work presents how the geometry of an LDR is leveraged along with optimized shield profiles and materials to design efficient neutron shields for high performance LDRs.

Performance Comparison of CSG and CAD-based Geometries in Monte Carlo Transport Simulations Using OpenMC

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The usual method for Monte Carlo transport geometry description is constructive solid geometry (CSG), but this approach can be challenging and error-prone, especially for complex models like fusion power devices since CSG is not a natural geometry representation for engineering analysis. For such geometries, the open-source transport code OpenMC supports Direct Accelerated Geometry Monte Carlo (DAGMC) for representing CAD-based models. Traditionally, the DAGMC workflow relies on Coreform Cubit for surface tessellation, with results stored in the Mesh Oriented DatABase (MOAB) as a collection of triangles with their associated connectivity and topology. In this study, we demonstrate a streamlined CAD-to-simulation workflow utilizing the Attila4MC contiguous mesh generator and the PyDAGMC Python interface to assign metadata to Attila's mesh for transport in OpenMC. The Attila-PyDAGMC workflow provides a relatively more straightforward method of generating discontiguous mesh models than Cubit, with the advantage that the resulting H5M file can be utilized as both an unstructured mesh for tallies and a DAGMC model for particle tracking.

The OpenMC Fusion Benchmarks GitHub repository is an open-source platform for benchmarking OpenMC against fusion experiments available in the Shielding Integral Benchmark Archive and Database (SINBAD). Benchmarks are typically defined and archived using CSG. With recent improvements in CSG-to-CAD workflows, the FNG HCPB Tritium Breeder Module Mock-up CSG model is converted to CAD geometry, aiming to standardize the CAD format within the repository. To this end, we analyze performance differences between CSG models, DAGMC models from the traditional Cubit workflow, and contiguous and discontiguous mesh-based DAGMC models from the Attila4MC-PyDAGMC workflow.

A major drawback of simulations using CAD-based geometries is the higher computational cost of ray tracing in DAGMC compared to CSG. To address this, we introduce mixed-precision ray tracing using Intel's Embree ray tracing kernel, which highlights performance improvements and advances in CAD-based simulations.

Increasing the current capacity of the Wendelstein 7-X In-Vessel Control Coils Power Supplies for short pulse operation

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The magnet system of the stellarator fusion device Wendelstein 7-X (W7-X) is composed of three different groups of coil systems. Beside the main magnetic field, which is created by a superconducting magnet system there are by two sets of normal conducting coil groups, the trim coils positioned outside of the cryostat and the control coils inside the plasma vessel.
The control coil system consists of ten 3D shaped coils with dimensions of 1.8x0.3x0.2 meters each), their power supplies, two separate cooling systems, high current feeds and an autonomous remote control system. The design of the ten individual power supplies is based on four-quadrant current converters using Mosfet-Transistors. They provide individual bipolar DC currents of up to 2.5 kA in steady state operation and a superposed common AC current of up to 625 A of low frequencies for each coil. The magnetic field created by the control coil system allows for the correction of error fields, to influence the islands at the plasma boundary and for the sweeping of the separatrix, e.g. the point of the largest power position, across the divertor.
After the operation phase (OP) 2.1 in 2023, the necessity for a larger influence on the island size and positioning by the control coils using higher DC currents above 2.5 kA became evident.
This paper describes the calculations, simulations and tests to evaluate the feasibility and the measures that have been taken to increase the current output of the power supply system from 2.5 kA to 3.5 kA for short pulse operation as well as the first results during the OP2.3 campaign of W7-X.

Multiscale Tritium Transport Modeling in Tritium-Breeder Cermets Using TMAP8

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Due to the low natural abundance and short half-life (12.3 years) of tritium, fusion energy systems need to replenish the burned and dissipated tritium inventory from the tritium-breeding blanket to be sustainable. The tritium-breeding blanket produce tritium when bombarded by high-energy neutrons from Deuterium-Tritium fusion reaction in the vacuum chamber. Lithium (Li)-based materials are leading candidates for such blanket due to their ability to produce tritium via nuclear reactions. Li-based ceramics are one of the most promising technologies for breeder blanket materials. Unfortunately, current materials suffer from irradiation-induced embrittlement, and adding allowing elements lead to reduced tritium production efficiency.

To address these challenges, we develop a multiscale tritium transport model to accelerate the design of a novel tritium-breeder cermet, leveraging the advanced manufacturing capability at Idaho National Laboratory, with aims at optimizing mechanical integrity and tritium generation and extraction. The model integrates mesoscale simulations with engineering-scale analysis conducted in TMAP8 (Tritium Migration Analysis Program, version 8). TMAP8 is a Multiphysics Object-Oriented Simulation Environment (MOOSE) framework -based, open-source, Nuclear Quality Assurance, Level 1 (NQA-1) compliant application designed to provide cutting-edge multiscale capabilities for tritium transport and fuel cycle modeling. The research explores the influence of key material characteristics, including the fraction of ceramics and metals phases and ceramic microstructure on tritium transport within cermets. This study aims to link cermet microstructure and tritium performance.

The presentation will detail the modeling and results, highlighting microstructural effects on tritium transport. The findings underscore the potential of the cermets to improve tritium production efficiency and enhance the safety and reliability of fusion energy systems.

MHD pressure drop in a Heat Pipe Liquid Channel for Fusion Reactors

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In our previous study (Jung et al., Fusion Sci. Technol., 2025), we proposed a heat pipe for cooling radio frequency (RF) antennas in fusion reactors as an alternative to forced convection cooling, which may pose additional efficiency and reliability challenges. A heat pipe is a widely-used passive cooling device that utilizes the latent heat of vaporization and the surface tension of the wick, operating without moving parts or forced flow. Considering the high temperature of plasma-facing components near 1000 K, liquid metals are recommended as the working fluid. However, the strong magnetic fields inherent in magnetic confinement fusion reactors additionally impose magnetohydrodynamic (MHD) effects on electrically conducting liquid metals. The MHD pressure drop along the liquid channel is known to significantly impair the heat transfer of the heat pipe (Carlson and Hoffman, J. Heat Transf., 1972, Matthews et al., Fusion Eng. Des., 2019), with the corresponding capillary limit under the MHD effect determining its performance. Hence, a detailed analysis on the MHD pressure drop is conducted for a rectangular channel with insulated and conducting walls. The liquid metal flow in the heat pipe channel can be described as a locally fully developed MHD flow, as typical Hartmann numbers are high, on the order of 1,000, while the flow remains laminar along with a negligible magnetic Reynolds number. The fully developed MHD pressure drops are investigated by varying the aspect ratio and the orientation of the applied perpendicular magnetic field, accounting for the potential effects of a 3-D geometry. The square channel shows minimal variation in the MHD pressure drop with the magnetic field orientation, making it a desirable geometry in terms of uniform mass flow rate and heat flux distribution. In addition, the test plan on liquid metal channels in a large bore magnet is proposed to validate MHD pressure drop and the heat pipe performance in various configurations.

ITER Low-Field Side Reflectometer (LFSR) Diagnostic Gaussian Telescope Design and Testing

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Burning plasma devices, such as ITER and future electricity-generating fusion facilities, impose new engineering challenges for fusion diagnostics that diverge substantially from prior experience. ITER diagnostics include various reflectometry systems, one of which is the Low Field Side Reflectometer (LFSR) that uses the frequency-dependent phase shift between a launched and reflected microwave to determine the plasma edge electron density. A broadband microwave beam is launched from the Diagnostic Hall and travels through ex-vessel regions (Gallery, Port Cell, & Interspace) via lengthy transmission lines (TLs) constructed of aluminum or stainless-steel circular tubing. The signal then reaches an in-vessel plasma facing antenna where it probes the plasma and is then reflected back to the Diagnostic Hall for processing. The overarching technical challenge is maintaining the required TL alignment to achieve acceptable reflected signal quality considering the potential misalignment tolerances resulting from the span of TLs containing several bends, mirror assemblies, secondary windows, and airgaps along with the non-uniform expansion of in- and ex-vessel components that occurs due to cyclic thermal loading. A Gaussian Telescope (GT) mechanism has been designed as a mechanical quasi-optical coupling device to accurately transmit the signal from the in-vessel antenna to the ex-vessel system under the temperature induced differential motion conditions. The GT uniquely integrates the physics and mechanical engineering aspects of plasma fusion diagnostic systems and has undergone multiple iterations of prototyping and design refinement to achieve desired transmission characteristics. The poster presentation aims to provide the technical requirements, engineering approach, and microwave test results of the GT design to share lessons learned and assist in the design of diagnostics in future integrated fusion facilities based on burning plasmas.

Liquid Nitrogen Superconducting Test Facility Cryostat Design Evaluations at the MIT Plasma Science and Fusion Center

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The Liquid Nitrogen Superconducting Test Facility at the MIT Plasma Science and Fusion Center (PSFC) provides a versatile and efficient environment for testing superconducting components, particularly for high-temperature superconductors (HTS) including tapes, cables, and coils. The facility, operational since 2020, plays a critical role in quality assurance for components such as the SPARC Toroidal Field Model Coil (TFMC) and Central Solenoid Model Coil (CSMC), and is supporting the development of HTS manufacturing processes. The test facility features a range of cryostats, with capacities from 20 to 4000 liters, and is designed with flexibility to accommodate evolving test requirements. This poster will evaluate how cryostat design impacts testing performance, including configurations with both silicon fiberglass linings and G-10 inner shell protection. These designs are assessed for their impact on liquid nitrogen usage and structural integrity.

The test setup includes a scalable 16 kA, 10 V power supply, and fast data acquisition system that captures data from up to 96 voltage taps, temperature sensors, and current sensors in real-time. The integration of a flow sensor into the LN2 setup allows for precise measurement of LN2 usage during filling, and also facilitates billing calculations based on actual consumption. Estimates of LN2 usage per hour are provided, allowing for an accurate understanding of operational costs and thermal performance. Data from temperature and magnetic field sensors, including RTDs, Cernox, Hall sensors, and fiber optic current sensors, are analyzed to optimize testing protocols and materials performance. The facility's intuitive control system enables dynamic adjustments to current ramp waveforms and real-time reprogramming during tests.

A recent example of the facility's capabilities is the CSMC experiments of the eddy current AC losses analysis of PIT-VIPER masked joint HTS cable, representative of a sub-assembly for the SPARC magnet, where data from voltage taps were used to characterize the sample and compare with predicted values. The facility's modularity, advanced measurement capabilities, and cryostat design innovations ensure its continued relevance for superconducting material testing and development.

A radiation hard charge amplifier for neutron diagnostics in nuclear fusion

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Neutron diagnostic systems are essential for the successful operation of nuclear fusion reactors. These systems must provide accurate measurements of neutron flux and neutron energy distribution while withstanding the extreme radiation environment inherent to fusion reactors. A critical component of such diagnostic systems is the analogue amplifier. To ensure high energy resolution, the charge amplifier must be placed close to the detector to minimise electronic noise and signal degradation. However, this placement exposes the amplifier to high neutron fluxes, necessitating a thorough understanding of its radiation tolerance and operational lifetime.

CIVIDEC's Cx-Mini is a small footprint ultra-low-noise charge sensitive spectroscopic preamplifier specifically designed for high-count-rate neutron measurements. With a Full Width at Half Maximum (FWHM) of 180 ns and gain of 3.5 mV/fC, the Cx-Mini delivers excellent time resolution and high-resolution neutron energy measurements for Deuterium-Deuterium and Deuterium-Tritium fusion plasmas.

In this study, five Cx-Mini amplifiers were irradiated at the TRIGA Mark-II fission reactor in Vienna. The amplifiers were exposed to 1 MeV-equivalent neutron fluences ranging from 1e12 up to 5e15 neutrons/cm2. To eliminate the impact of the neutron energy distribution, the 1 MeV-equivalent neutron flux for silicon was used, following the Non-Ionizing Energy Loss (NIEL) scaling hypothesis. This represents the neutron flux of monoenergetic 1 MeV neutrons that would generate the same damage as the real neutron energy distribution, enabling lifetime estimations for a wide range of neutron environments.

The amplifiers were electronically calibrated before and after irradiation to determine their gain, noise and pulse characteristics and evaluate the performance stability as function of dose. These measurements aim to identify the radiation-induced degradation of the preamplifier performance and establish lifetime limits for the Cx-Mini. The results provide critical insights into the design of radiation-hard electronics for future neutron diagnostics for nuclear fusion.

Radioactive Inventories of Fusion power plants: Challenges and Perspectives

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

The assessment of worker and public safety impacts from the deployment of fusion power plants (FPPs) requires evaluation of the radiological source term (i.e. the amount and nature of the radioactive materials) that could be released during accidents or security incidents. This requires analysing the radioactive inventory within the FPP and determining the fraction that inventory could be mobilised under adverse conditions. Most commercial FPP concepts currently under development are in the early design stage, have limited data on radioactive inventories, and state that their current designs are expected to have radioactive inventories one order of magnitude less than ITER, proposals for EU DEMO, and other previous FPP design studies. This work provides estimates for FPP hazard inventories using publicly available information and the associated radiological source terms. Tritium is identified as the primary radiological hazard across FPP designs, but there are significant differences in both tritium inventories and in accident mobilisation due to differing design choices. Consequently, establishing a consistent relationship between FPP technologies and radioactive inventories is challenging. The study highlights the need for standardised methodologies to accurately assess and compare radioactive material inventories in FPPs. It also emphasises the scarcity of safety assessment data for many developing FPP concepts, underscoring the importance of integrating safety considerations in early design stages.

Initiatives to grow new innovate talent to enable fusion energy (IGNITE Fusion Energy)

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The rapid advancements in fusion energy research underlines an urgent need for a strong workforce capable of driving future fusion engineering and technology developments. These advancements, coupled with the needs for commercial fusion energy, highlight the pressing need to grow and sustainably develop the talent pipeline in fusion engineering and technology. To address this critical demand, we have launched the “Initiatives to Grow New Innovate Talent to Enable Fusion Energy (IGNITE Fusion Energy)” project, funded by the Department of Energy’s Reaching a New Energy Sciences Workforce (RENEW) program. This project aims to leverage the collective strengths of six academic institutions (Tennessee Tech University, Tennessee State University, Tuskegee University, Southern Adventist University, Missouri University of Science and Technology, and University of Tennessee – Knoxville), the Oak Ridge National Laboratory (ORNL), and more than ten private fusion companies to create workforce training initiatives and enhance curriculum development, thereby preparing a new generation of researchers and technicians for careers in fusion engineering and technology.

This initiative is developed around the following three objectives. (1) We are establishing a student mentoring program to involve undergraduate and graduate students through consecutive summer internships at ORNL and a private fusion company, where they will gain hands-on experience with cutting-edge research and commercial applications of fusion technology. Students will also benefit from continued mentorship in the form of career guidance and professional development during academic semesters. (2) We are collectively developing a series of new courses and special-topics modules to be implemented at participating universities, which will later be shared publicly with the broader academic community. This curriculum development effort will be guided by private fusion companies and national lab leaders, reflecting the needs and recent advancements in fusion engineering and technology. (3) To promote the sustainability of the project efforts, we will create an inaugural entrepreneurship and innovation focused bootcamp – the Fusion Innovation Bootcamp – designed for sustained training and participation of students in fusion engineering. Student trainees will engage with fusion startup professionals, national lab researchers, and university faculties in a dynamic curriculum featuring lectures, panels, hands-on sessions, and pitch presentations.

The integration of student mentoring, curriculum development, and bootcamp engagement ensures that the training and involvement of participants is sustained beyond the duration of this RENEW project and into the future of fusion engineering and technology. This RENEW project aims to sustainably improve workforce conversion and retention in the fusion industry, enrich fusion engineering curricula across academia with vital course materials, and establish a sustainable talent pipeline from academic institutions to the public and private fusion sectors.

Automatic Creation of a Nuclear Fusion Energy Knowledge Base for Effective Elicitation and Retrieval of Information

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The achievement of nuclear fusion power presents numerous challenges that extend beyond the inherent complexities of nuclear physics and require multidisciplinary approaches. As the area of research continues to expand, the need for an efficient system of data management and knowledge retrieval has become increasingly critical. Addressing this challenge requires not only improving data accessibility but also enabling seamless interaction between human understanding and machine processing.

An interconnected system for online data, that can bridge this gap, holds significant appeal to many disciplines beyond nuclear fusion energy. To address this need, new data models have been developed based on graph structures and ontology frameworks, that can be read both by humans and machines. These two approaches represent complementary data models that, when used together, can enhance the accuracy and richness of the final Knowledge Base (KB).

The development of a KB of fusion energy, which organizes domain-specific information into structured categories and representations, will accelerate the development of more ''FAIR'' data sharing, disseminating and referencing. Over time, such a system will foster collaborations between all partners in a fusion supply chain, helping to de-risk the design, development and construction of fusion power plants, while also reducing the barriers to understanding between field experts, funding agencies and policy makers.

In this work, we present an automatic approach to generate a graph-based KB of nuclear fusion energy starting from a large corpus of scientific documents and by leveraging the inference power of pre-trained Large Language Models (LLMs). We will also present the development of a Knowledge Graph Retrieval-Augmented Generation (KG-RAG) machine. This combines the advanced linguistic skills of modern generative AI with domain-specific knowledge extracted from previously unseen sources. This KG-RAG enables the effective retrieval of domain-specific information and the generation of accurate answers to user queries while minimizing model hallucinations and the production of plausible but fictitious responses. In the end, we discuss the challenges encountered throughout the project, including data scoping, knowledge extraction, and managing noise in the retrieval and generation processes.

Safety Approach, Requirements, and Preliminary Analyses for the Design of the VNS Facility

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Volumetric Neutron Source is a nuclear plasma device that serves as a 14 MeV neutron for testing and qualification of the Breeding Blanket (BB) to be run in parallel to both ITER operation and the DEMO design process. VNS is expected to be small, with an associated low power, a tritium consumption of around 1.5 kg per full power year, and with an easy access to the fusion core. The safety principles and approach are designed to be proportionate to the safety and environmental risks associated with the given performance levels. The foundational safety principles employed in the pursuit of safety are Defence in Depth, ALARA, and passive safety. These principles collectively serve to mitigate risks and enhance the robustness of safety systems in complex environments. At the top level, three Fundamental Safety Functions are defined for VNS: confinement of hazardous materials, reduction of workers' exposure and limitation of the environmental impact. Being the goal the definition of supporting safety functions, the use of safety analyses are mandatory to assess that the VNS design aligns wih it. As part of the safety assessments, preliminary accidental analysis studies have been conducted to support the assessment of the dynamic response and implications related to accidental conditions. The accidental analyses have been performed through the MELCOR 1.86 system code. The MELCOR model comprises the Shielding Blanket (SB), a water-cooled Test Blanket System, modelled as circuits with water as working fluid, as well as the Vacuum Vessel (VV) and its extensions. The basic modeling has been performed using Control Volume Hydrodynamics (CVH) nodes, Flow Links (FL) for the flow paths, valves and pumps. Different Heat Structures have been used in order so represent solid components and working fluids different from water. To guarantee the completeness of these analyses, it is crucial that potential accident event sequences are thoroughly identified. This requires the identification of Postulated Initiating Events (PIEs). Two events have been evaluated as reference events: an in-vessel Loss Of Coolant Accident (LOCA) from BB and SB with Loss Of Offsite Power (LOOP), and an ex-vessel LOCA from BB with SB loop with LOOP. In the context of in-vessel events, the VV Pressure Suppression System (VVPSS) served as an effective mitigation mechanism, significantly limiting radiological releases and effectively confining contaminants during the accidental scenario. About the ex-vessel event, the analysis demonstrates that the facility's safety systems, like pressure relief flaps, effectively mitigate pressure buildup and limit radiological releases during the ex-vessel LOCA scenario. However, the water flashes into steam that invade the galleries close to the Upper Pipe Chase (UPC). The UPC results in rising in temperature which causes to the stop of the normal operation of the detitriation system, entailing in firstly a release of tritiated water (HTO) into the tokamak building and than leakage toward the environment. Safety analyses are conducted to ensure that the design effectively provides all required safety functions and meets all safety requirements.

Effects of Surface Geometry on Wetting With Liquid Lithium

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Several liquid metal concepts have been proposed for use as first wall components in magnetic confinement fusion reactors. One prominent metal currently under investigation is lithium. For a liquid metal layer to protect other reactor components the surface of these components must be fully wetted with the metal. Recent research has clearly demonstrated that a simple flat plate is not sufficient to encourage wetting of flowing lithium at temperatures low enough to prevent significant lithium evaporation.

The Test Housing for Advanced Liquid Aloy Surface Studies and Applications (THALASSA, formerly known as MEME) is being used to follow up on previous research at the Center for Plasma Material Interactions (CPMI) showing that a porous surface encourages more wetting with lithium than a smooth surface. To test this, a continuous flow liquid lithium loop is constructed within THALASSA and connected to a modular plate holder. The modular nature of this plate holder allows for different plates of various surface geometries to quickly be installed and tested. Previous work using this system has shown the importance of the placement of loop components in relation to the plate and how the direction from which lithium enters the plate affects how the plate is wetted.

Ongoing research aims to provide evidence that a either a porous or channeled surface, or some combination thereof can be wetted sufficiently to provide an uninterrupted protective layer to the wall of a divertor region while maintaining lithium at low enough temperatures that evaporation does not significantly contaminate the core plasma or lead to material loss of the working fluid of the loop. The next phase of research will use a Helmholtz coil to simulate a toroidal magnetic field to analyze the behavior of the flowing lithium in the presence of such a field.

Application of neutron diagnostics for fusion power measurement in the SPARC tokamak

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This paper discusses the neutron diagnostics of the SPARC tokamak and shows how it will establish multiple independent and redundant estimates of the total fusion power (Pfus). SPARC is a compact, high field, DT tokamak, under construction and assembly by Commonwealth Fusion Systems (CFS) in Devens, MA. One of its mission goals is to demonstrate a fusion breakeven, i.e., the fusion gain (Qfus)>1 as early as possible. SPARC is designed to approach Pfus~140 MW, Qfus~11, and DT neutron yield rate (dYn/dt) ~ 5e19 n/s. The latter is measured with varying degrees of temporal, spatial and energy resolutions using the four distinct neutron systems of SPARC; namely, neutron flux monitors, neutron activation system, radial neutron camera, and a magnetic proton recoil spectrometer. The target accuracy on Pfus is set at <±10%, and design targets and early sensitivity studies show that the majority of the systematic uncertainties can be well-contained under ±10%. To achieve that, a high intensity (>1e8 n/s) DT neutron generator will be used for an in-situ calibration of the SPARC neutron diagnostics during the commissioning phase. Absolute calibration is supported by high fidelity neutronics model(s) of the device and its surroundings. SPARC further plans to use the spectrometer (with its ab-initio calibration) and camera systems to provide absolute neutron yield, an approach that has high potential cost and time savings. With an overview of the state of the SPARC neutron systems' designs and prototyping, the paper presents preparations per system to build workflows for estimating neutron yield rate and Pfus and their respective uncertainties.

High frequency transformer model in high magnetic field based on neural network

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As an important component of switching power supply, high-frequency transformer plays the role of voltage conversion, electrical isolation, energy conversion, etc., and is widely used in the power supply of auxiliary system of Tokamak devices. However, the transformer is easy to be disturbed by the strong stray magnetic field generated by the Tokamak device in the surrounding space, which leads to the change of its operating characteristics and affects the output performance of the switching power supply. The accurate model of high frequency transformer in strong magnetic field is the premise of power supply immunity design. This paper proposes a transformer model in strong magnetic field based on finite element simulation and neural network. The finite element simulation is used to calculate the field distribution of the transformer under different strong magnetic field conditions, and the key parameters that can describe the performance of the transformer are extracted. The relationship model between the key performance parameters of the transformer and the strong magnetic field parameters is established through the neural network and brought into the simulation model of the power supply. Finally, the simulation results of the power supply are compared with the test results of the real power supply in the strong magnetic field to prove the accuracy of the proposed scheme. This paper provides a novel idea for the establishment of magnetic element model in strong magnetic field.

AtomCraft: Training the fusion workforce by delivering the first tokamak entirely designed, built and operated by undergraduate university students

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Sustaining the rapid growth of the fusion industry will require a proportionate increase in the supply of skilled workers who can engineer, manufacture and deliver critical components for nuclear fusion machines. Few universities worldwide provide specialized education in fusion energy, and those focus mostly on physics rather than engineering. Thus, the global challenge of engineering workforce gap will be felt particularly acutely in the fusion industry in the decade to come. Here we present a novel approach to massively accelerate and expand university training of fusion workforce through project-based learning.

Launched in 2024 at UNSW Sydney, “AtomCraft” is a world-first university engineering course that aims to deliver the first tokamak entirely design, built and operated by undergraduate students. The students are exposed to real-world engineering practices and challenges, with a focus is on the design and construction phases. This contrasts with other small-scale university tokamaks, which focus on providing operational access to an existing machine.

The project will deliver a new prototype every 3 years (not all tokamaks), each addressing specific design constraints, informed by our industry partners. The first machine is a conventional small-scale tokamak with a major radius of 0.3 m, aspect ratio of ~4 and targeted toroidal field of 0.1 T. The main technical goal is to demonstrate a 100 ms pulse with ECRH-assisted plasma breakdown and a peak plasma current of 5 kA.

Through AtomCraft, students are equipped with foundational knowledge on how to design and integrate tokamak subsystems, in close collaboration with industry partners. Students are developing skills in: vacuum engineering, RF engineering, power electronics, plasma diagnostics and EM and plasma simulation. We will outline our current progress and lessons learned thus far. The design methodology underpinning the AtomCraft tokamak is also presented, and is intended as a reference

Integrated Containment and Confinement Strategies for Tritium Safety and Environmental Protection in the H3AT Facility

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Safe tritium management in fuel cycle facilities requires robust containment and confinement strategies to minimise environmental impact and ensure worker safety. The H3AT (Hydrogen-3 Advanced Technology) facility, UKAEA's tritium processing facility currently under design, employs a multi-layered approach to tritium confinement. Primary containment is achieved through high1 integrity, low-leakage components within the process pipework, secondary containment is provided by gloveboxes, and an Atmosphere Detritiation System (ADS) serves as the final protective layer. Achieving effective tritium confinement in the facility requires overcoming the challenge of integrating the multiple layers into a cohesive containment and confinement system, particularly maintaining stable pressure control within gloveboxes under varying operational conditions and ensuring seamless coordination across containment mechanisms.

The H3AT facility's design demonstrates how integrating gloveboxes equipped with precise pressure control mechanisms and the ADS through a common manifold ensures effective tritium confinement and robust protection against environmental releases. Secondary containment systems have often been prominent safety measures in tritium-handling facilities, providing an essential line of defence against potential releases [1-2]. In contrast, the H3AT facility's multi-layered approach evenly distributes risk reduction across all containment layers, minimising reliance on any single system. The ADS not only serves as the final protective layer by removing and recovering tritium from gaseous effluents to reduce releases to ALARP levels but also maintains sub-atmospheric pressure in gloveboxes. Additionally, the pressure control system within the gloveboxes ensures operational stability by responding to deviations from permissible pressure ranges, enhancing safety and containment reliability compared to traditional single-barrier approaches.

This integrated approach represents a shift toward holistic tritium safety designs in fusion facilities, emphasising the interplay between active and passive systems to enhance both operational stability and environmental protection. Balancing these containment strategies, the H3AT facility aligns with broader industry goals of achieving ALARP for tritium emissions while maintaining operational efficiency. The H3AT facility's design process contributes valuable insights to the evolving standards for tritium management in fusion and other advanced nuclear technologies. By addressing current challenges, it sets the stage for developing more resilient and adaptable containment systems, with lessons directly applicable to future large-scale tritium handling facilities, such as ITER and STEP. Once operational, the H3AT facility will generate critical data and operational experience, further refining strategies for tritium containment and confinement.

Keywords: Containment, confinement, detritiation, glovebox, pressure control.

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The UK Journey: Advancing a Proportionate and Enabling Fusion Regulatory Framework with Evidence-Based Decisions

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The development of fusion energy presents unique challenges, requiring regulatory frameworks that are both proportionate and forward-thinking. These frameworks must ensure safety, security, and environmental protection, foster public confidence, and support technological advancement. This work explores the UK's journey toward establishing a regulatory framework that enables public and private fusion companies, focusing on the integration of scientific assessments into policy decisions, and the importance of collaborative stakeholder engagement.

Central to the UK's approach is the balance between fostering innovation and proportionate regulation whilst maintaining safety standards. The UKAEA Technology Report for fusion was created leveraging publicly available safety studies on early fusion power plant concepts. Within this the level of risk that may be associated with hypothetical accident scenarios at large fusion power plant were shown to be low. The UK has aimed to address concerns from both the public and industry stakeholders via Government Consultation on its proposals for a regulatory framework for fusion, ensuring that regulation keeps pace with technological developments. Collaborative dialogue among policymakers, researchers, and industry representatives has also been instrumental in shaping strategies that align with global best practices, positioning the UK as one of the leaders in the fusion regulation.

Key themes of this process include the importance of proportionate regulation, and the necessity of maintaining public trust through transparent and evidence-based decision-making. This work highlights how the UK's experience could contribute to international co-operation in establishing a harmonised regulatory framework for fusion and provide a pathway for navigating the complexities of fusion regulation. Such an approach not only facilitates technological innovation but also ensures that safety and public confidence remain at the forefront of the fusion journey.

Homogenized Viscoplastic Material Model for a Dual Cooled Lead Lithium Blanket

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A non-linear multiphysics component scale finite element blanket model and homogenized viscoplastic material model was developed for modeling irradiation and cavity swelling creep in the dual cooled lead lithium (DCLL) blanket. Since the constitutive behavior of a material and the evolution of the voids within are highly coupled process, a theorem proposed by Bishop and Hill was employed to match the dissipative potential and ensure accurate constitutive behavior of the blanket material system. The viscoplastic model is based on Gurson-Tvergaard-Needleman implementation that is widely used for modeling damage evolution in ductile materials. A separate model that correlates changes in neutron damage and operational temperature to void swelling of F82H steel at fusion relevant temperatures is provided to supply the volume fraction of the voids. The results of the homogenized stress and the stress induced evolution of the cavities in the first wall, back wall, separation plate of the blanket for a period of 3000hrs are presented. Results are compared with another Leblond-Perrin-Suquet based viscoplastic model for the DCLL blanket.

Divertor Monoblock Multiphysics

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To study shared issues for ARC-based fusion energy system concepts, the design of outer fuel cycle (heat transfer system) components needs to be explored. In particular, the primary heat exchanger serves as an interface between the outer fuel cycle and power conversion systems while influencing the inner fuel cycle system. Therefore the primary heat exchanger(s) has to transfer the heat required for power generation that results in the heat exchanger having the largest surface area in the outer fuel cycle. The high amount of surface area also yields significant tritium mass transfer meaning the primary heat exchanger(s) impacts the inner fuel cycle. These attributes force design requirements and limitations on the primary heat exchanger to include tritium barriers to reduce tritium fuel losses and radiological risks. The heat exchangers are assumed to have molten salt (FLiBe) transferring heat to a solar (nitrate) salt. The solar salt is selected for the existing supply chain for concentrated solar power plants.

In this study, the design space of primary heat exchanger(s) are explored using design limits and figures of merit with respect to both inner and outer fuel cycles. The design limits and figures of merit are based on both fusion energy systems and cross-cutting molten salt reactor technologies due to inherent commonalities. The initial design limits include coolant and material temperatures, allowable pressure drop, localized flow velocity and maximum tritium losses. While the initial figures of merit includes coolant volume, component size, temperature drop, and manufacturability. With estimated design limits and figures of merit, the design space is determined using heat exchanger sizing calculations for shell and tube heat exchangers with and without heat transfer enhancements. The sizing calculations are done using pre-established methods used for any industrial heat exchanger coded in Python. Whereas, the design space is established using Pareto-front based multi-objective optimization connected to the sizing calculation code. The multi-objective optimization will involve balancing both design limits and figures of merit mentioned. Down-selected primary heat exchangers designs are presented with detailed knowledge gaps that will need to be addressed. These knowledge gaps will then be addressed in follow-on studies using both computational and experimental studies.

Preliminary design of the next-generation levitated dipole experiment Tahi

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Levitated dipole experiments have demonstrated successful plasma confinement and validated fundamental magnetospheric physics in laboratory settings. However, these experiments have lacked the heating power density required to achieve fusion-relevant plasma conditions leaving their performance as fusion devices yet to be experimentally investigated. Following the completion of OpenStar Technologies' first device "Junior", the next-generation machine "Tahi" is designed to achieve dense, hot thermal ion populations to investigate their performance in this plasma regime and provide the stepping stone necessary for the development of future higher-performance devices. Planned for construction by 2027, Tahi offers significant upgrades to previous levitated dipole experiments in its heating systems, magnet performance, and plasma diagnostics. This work provides an overview of the current design status of these systems, predictions of machine performance, and details how these key systems will enable completion of Tahi's physics goals.

Modeling the Lobo Lead Loop in MOOSE: Development of a Summer School Module

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To meet the world's growing energy demands, fusion energy technologies are rapidly growing. With the increasing demands of the fusion industry, there is a need for growth in the workforce. In a 2020 report from the Fusion Energy Sciences Advisory Committee (FESAC), it was identified that the fusion and plasma communities have low participation of women and underrepresented minorities. FESAC encourages programs to broaden recruitment pools to increase opportunities for these underrepresented groups in national laboratories, universities, and private industry. To help recruit and ultimately develop a more diverse workforce in fusion research, the University of New Mexico and the University of Tennessee-Knoxville are working together to create a widely available liquid metal summer school. Within that school, we aim to develop an educational module focused on modeling and simulation with the open-source MOOSE application [1]. While NCRC applications like RELAP provide high-fidelity models for these concepts, MOOSE allows for increased accessibility of these capabilities and a larger variety of multi-physics modules. Additionally, it allows for the introduction of these modeling capacities earlier in students' careers.
In the presented work, one training module is discussed in depth. This module is meant to teach students to model the University of New Mexico's Lobo lead loop in MOOSE. The Lobo lead loop supports numerous studies on molten lead interactions, including corrosion and erosion of structural materials, instrumentation, radioisotope retention, and thermal hydraulics. Modeling the loop in MOOSE allows for the computational understanding of thermal hydraulics under a wide variety of boundary conditions. While this represents a specific application of MOOSE, the course will help students build broad foundational skills in modeling and simulation. We will begin by outlining the learning objectives for the module. A broad overview of the course contents and delivery methods will be outlined. Finally, lessons learned from the first administration of the course will be discussed.
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LabVIEW based Application Software for the ITER Diagnostic Residual Gas Analyzer Prototype

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The Diagnostic Residual Gas Analyzer (DRGA) is a multi-sensor diagnostic system for the ITER project and is capable of measuring fusion gas compositions with very high resolution [1]. The DRGA can resolve isotopic compositions of hydrogen and helium as well as other heavier elements and compounds [2]. The DRGA analyzes the composition of gases in a vacuum chamber with mass spectrometry and optical gas analysis techniques. Measurements from residual gas analyzers (RGA, the sensors providing the mass spectrometry), Charge Coupled Devices (CCD) cameras (used for optical spectroscopy detection), and Pirani/Cold cathode gauges are combined to calculate partial pressure of various fusion gas species. The Joint European Torus (JET) had a functionally equivalent system for sub-divertor neutral gas analysis (JET internal name KT5) [3]. JET is closest to the ITER tokamak in terms of operating parameters and size, thus providing important validation of the approach for ITER, including experience with DT plasmas [2]. Oak Ridge National Laboratory (ORNL) has developed a DRGA prototype to validate various instruments, both commercially available and specialized. The prototype can provide ITER like pressure conditions and gas species [4, 1]. ORNL has developed a LabVIEW™-based software to interface with various instruments on the prototype, to explore how to overcome dependence on vendor-provided, standalone solutions. The software acquires raw data from various instruments simultaneously and processes them to produce useful physics data. This software will help validate the hardware as per the ITER requirements of measuring various gas species within specified constraints. This paper will explain the software architecture and operating states of the DRGA in detail. The operation state machine orchestration is designed similar to ITER operating modes. The Graphical User Interface (GUI) is designed considering the DRGA physics needs for visualizing gas species concentrations in profile plots and trend charts for individual masses. The software helps identify and decode the communication protocol of commercial instruments to acquire raw data. Understanding of the communication protocol will help to develop an Experimental Physis Control system (EPICS) based software for ITER machine in future.

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Novel Heat Treatment for Reduced Activation Ferritic/Martensitic Steel Welds

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Structural components in a fusion reactor must withstand the diverse range of demands placed on the components, including 14 MeV neutron irradiation, He exposure and temperatures up to 650°C. Joints between structural materials must maintain integrity during operations, but residual stresses that are left after welding can cause them to fail unpredictably. The main candidate structural materials for breeder blanket module designs are reduced activation ferritic/martensitic (RAFM) steels because of their reduced swelling, high temperature capability and the faster decay of induced radioactivity following service [1]. RAFM steels, such as Eurofer97, will be used in the UK Atomic Energy Authority's (UKAEA) STEP (Spherical Tokamak for Energy Production) programme's breeder blanket design, which involves a complex assembly process; a joining strategy must be developed using a range of techniques.

Electron beam and laser welding are promising techniques for RAFM steels, with little distortion and high pass speed compared to gas tungsten arc welding, for example. RAFM steels are sensitive to rapid temperature changes due to the solid-state phase transformations that occur in the fusion zone, which cause residual stresses to be generated in the weld. Post-weld heat treatments have been used to reduce these stresses but cannot completely recover the original parent microstructure [2].

Several techniques have been used to measure the residual stress across Eurofer97 laser welds, such as neutron diffraction [3] and Bragg edge imaging [4] before and after a standard post-weld heat treatment. Whilst they show that the residual stresses have decreased, concentrations of residual stress remain that can cause engineering complications such as buckling or cracking. In line with UKAEA's NEURONE (NEUtron iRradiatiOn of advaNced stEels) programme, of which joining is a key area of research, this work explores the application of novel post-weld heat treatments to Eurofer97 electron beam welds. These will be analysed ex-situ and in-situ using neutron diffraction cross-validated with the contour method, a destructive technique developed by Prime at Los Alamos National Laboratory [5]. Characterisation using EBSD is used to reveal the changes in microstructure across the fusion and heat affected zones of the weld.

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Optimization of precipitate segmentation through linear genetic programming of image processing

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

Neutron flux in fusion reactors causes voids and swelling in materials—and creates nuclear waste in materials susceptible to neutron activation of long-lived isotopes. Current additive manufactured niobium-based copper alloys such as GRCop-84 have high yield-strength due to precipitate formations, which prevent swelling. Neutron activation of niobium produces long-lived isotopes; new reduced-activation copper alloys with similar yield-strength are needed. The yield-strength of a material can be estimated through analysis of the material's precipitate size distribution from electron micrographs of focused-ion-beam (FIB) milled cross-sections. Current analysis relies on hand annotation due to varying contrast, noise, and image artifacts present in micrographs, slowing iteration speed. We present a filtering and segmentation algorithm for detecting precipitates in FIB cross-section micrographs, optimized using linear genetic programming (LGP), which accounts for the various artifacts. Our LGP environment uses a domain-specific language for image processing centered around iterative processing of an input (folding), where we can reliably generate, mutate, represent, and execute filter pipelines. Our environment produces optimized human-interpretable MATLAB code representing an image filtering pipeline. Under ideal conditions—a population size of 60 and a maximum program length of 6 blocks—our system was able to find a near-human accuracy solution with an average evaluation error of 1.79% when comparing segmentations pixel-by-pixel to a human baseline. Our automation work enabled faster iteration cycles and furthered exploration of the material composition and processing space: our algorithm processes a 3.619 megapixel image in ~2 seconds on average. This ultimately enables convergence on strong, low-activation, precipitation hardened copper alloys for additive manufactured fusion reactor parts.

Helium enrichment and tritium burn efficiency in simulations of divertor plasmas

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The connection between projected fusion reactor performance and helium enrichment in the edge plasma is explored using the plasma edge code SOLPS-ITER. The first fusion power plants (FPPs) will use the deuterium-tritium (D-T) fusion cycle, and on-site production and recovery of tritium must sustain (or exceed) the fueling needs of the plant. Recently, it was proposed that the tritium usage in steady state, equilibrated fusion reactors could be characterized by a generic, dimensionless figure of merit: the “Tritium Burn Efficiency”, or “TBE” [1]. Since each D-T fusion reaction will produce one neutron (used to breed tritium) and one alpha particle (pumped out of the divertor as helium ash), this metric relates the rates of fuelling input to the core plasma, tritium burn, and helium exhaust to derive analytical expressions for the tritium throughput and global helium content. In this framing, the reactor performance is fundamentally dependent on the permitted helium gas fraction in the divertor and the effective pumping speeds of both helium and unburned hydrogenic fuel.

This study addresses the applicability of TBE for characterizing fusion devices through evaluation of helium transport and enrichment in the divertor plasma, in both existing experiments and then for next-step devices. Although existing tokamaks do not produce significant fusion-born alpha particles, these machines can still provide a reasonable test environment for studying helium exhaust and transport, where helium can be injected as an extrinsic impurity species. Existing impurity enrichment studies using helium performed on the DIII-D tokamak with induced scrape-off layer flows at various puffing/pumping speeds are modeled using SOLPS-ITER, matching reported experimental conditions [2]. Though tritium is not used in these experiments, since TBE is dependent only on quantities relating to helium in the edge plasma, the TBE parameter can still be used to interpret and predict the relationship between observed helium enrichment and fusion performance. For D-only plasmas, preliminary estimates of TBE from reported values in existing work suggest TBE of >5-10% should be achievable, but further analysis is needed to better understand the efficacy of TBE in the presence of the complex dynamics of the plasma edge. The simulations performed in this work provide a validated baseline for further physics studies on divertor helium enrichment. The experimental conditions for the existing discharges are held constant, and the influence of various processes (such as helium transport, recycling, and pumping speeds) on helium enrichment in the divertor are explored. The resulting impact of these factors on TBE is discussed.

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Overview of the SPARC Vacuum Cleanliness Program

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

The SPARC tokamak includes several separate and interconnected vacuum and high purity systems that need to meet strict and sometimes unique cleanliness and material compatibility requirements to ensure they meet functional requirements for plasma startup and operation, diagnostic systems function, cryogenic system performance, and tritium processing compatibility. There are many examples of poor performance, delays, and damage in the history of fusion due to contamination and debris left in the machine after assembly and maintenance. A vacuum cleanliness program has been implemented to mitigate risks caused by poor vacuum quality for each of the separate and connected vacuum volumes, establishing cleanliness requirements and material restrictions for manufacturing, inspection, and operation.
This report describes the technical cleanliness requirements that the integrated SPARC vacuum systems need to meet depending on the driving vacuum, plasma, and tritium compatibility requirements. We present a strategy for managing technical risk and validation methods through the use of guidance, thresholds, qualification tests, and integrated outgassing and leak rate budgets. A summary of the technical basis and impact assessment behind these driving requirements and the development of general impurity limits and outgassing budgets, material restrictions and tritium management strategies is presented. We describe the expected outgassing rates and system vacuum partial pressures from initial evacuation, bake, and cooldown of the device, and present example measurements of cleanliness and outgassing rates used to evaluate materials.

In-situ pellet presence diagnostic with thermal waves in DMS SPI injectors

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

Disruption Mitigation System Shattered Pellet Injector (DMS SPI) technology is emerging as a critical machine protection subsystem for larger tokamaks, preventing severe damage from plasma disruptions. A novel diagnostic approach employs thermal wave propagation within the injector barrel to provide in-situ assessment of this safety-critical system. By introducing a controlled thermal disturbance in the barrel and measuring temperatures at two distinct points, both the phase shift and amplitude variations of the resulting thermal waves can be tracked. From these measurements, the material's thermal diffusivity is derived, which allows the detection of pellet presence or absence in the barrel. This method offers a non-intrusive means of evaluating the readiness of DMS SPI components in future fusion reactors. Moreover, its capability to distinguish between loaded and unloaded states could significantly enhance reliability, ensuring that SPI pellets are deployed accurately when incipient plasma instabilities arise. Consequently, thermal wave analysis stands as a promising step toward more robust, in-situ monitoring of DMS SPI systems, ultimately contributing to the reliable operation of fusion devices.

Thermal-Hydraulic Modeling and Code Benchmark of the EU-DEMO Tokamak Building Under Accident Conditions

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One of the main objectives of the EU-DEMO plant is to demonstrate the feasibility of generating electricity from nuclear fusion while developing the technological solutions necessary for both plasma control and broader operational demands. However, the plant's significant radioactive inventory, intrinsic complexity, and high-enthalpy cooling loops demand a robust safety strategy to fully harness fusion power's environmental benefits. In this context, the Tokamak building—serving as the ultimate confinement barrier—must address critical challenges related to overpressurization and radioactive mass transport hazards. Under the EUROfusion Safety and Environment (SAE) work package, extensive safety studies are being conducted to guide design improvements, focusing on the thermal-hydraulic behavior of the confinement building under design-basis accident (DBA) and design extension condition (DEC) scenarios, as well as on source term mobilization.
This paper discusses the development of a thermal-hydraulic model of the EU-DEMO Tokamak building and, as part of a verification phase, presents a benchmark analysis employing MELCOR and GOTHIC codes. The main goal of this work is to estimate the ultimate pressurization of the different building areas and the potential radiological releases within the scope of a code-comparison approach. A preliminary model of the heat ventilation and air conditioning system and the vent detritiation system has been included to simulate the impact of dynamic confinement barriers during a loss-of-coolant accident scenario. Results suggest that building design improvements may be necessary to maintain pressure within safety limits; initial mitigation strategies are also explored to enhance the overall safety performance of the building.

"Photographing" a Fusion Kitty Hawk: Measuring Qp > 1 and Beyond on SPARC with a Spectrometric Neutron Camera

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Plasma energy breakeven $Q_p = P_{fus}/P_{aux} > 1$ has long been sought as evidence for fusion's feasibility as a source of energy. The SPARC tokamak, now under construction in Devens, MA by Commonwealth Fusion Systems (CFS) is predicted to not only achieve $Q_p > 1$ but enter the burning plasma regime $Q_p > 5$ producing at most 140 MW of DT fusion power or $\sim 5 \times 10^{19}$ neutrons per second. In order to verify this predicted performance, a suite of four neutron diagnostics is being designed and built for the machine. This includes a poloidal neutron camera which will be capable of resolving neutron emission in time, space, and energy, from which the neutron emissivity and ion temperature profiles can be derived. With an appropriate calibration, the fusion power can also be assessed. Neutron cameras have been fielded on many other magnetic confinement fusion devices (JET, TFTR, LHD, MAST-U) but the SPARC neutron camera will be the first to operate with energy-resolved detector units, enabling more accurate emissivity reconstructions and measurement of the ion temperature profile. This spectral capability is enabled by two modern detector technologies: single-crystal chemical vapor deposition diamonds and deuterated-xylene liquid organic scintillators. By combining these two detector technologies, the SPARC neutron camera will be capable of operating over three orders of magnitude for both deuterium-deuterium (2.45 MeV) and deuterium-tritium (14.1 MeV) neutron emission. In this presentation, we share details of the design of the spectrometric detector units, including results from a complete NCAM channel model assembled at CFS and MIT with custom detector prototypes. We also present synthetic tomographic reconstructions of both neutron emissivity and ion temperature for a range plasma scenarios and present a methodology for quantifying the uncertainty in neutron camera measurements.

This work is supported by Commonwealth Fusion Systems.

IQ demodulation with real-time transmission line compensation for ITER bolometer diagnostics

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The reference sensor for the ITER bolometer diagnostic is a miniaturized metal resistor bolometer, similar to other machines [1] . Resistive bolometers detect the temperature change in the absorber by a change of resistance in a metallic resistor placed underneath the absorber. Two absorbers, measurement and reference absorber, are combined within one bolometer channel while the reference absorber is shielded from plasma radiation. The resistors underneath both absorbers are combined in the electrical scheme of a Wheatstone bridge. This results in a very sensitive signal proportional to the incoming plasma radiation. The Wheatstone bridge is mounted on the vessel of the tokamak, whereas the acquisition system needs to be distant from the vessel due to possible radiation damage. Therefore, a transmission line, whose length is not negligible, connects the sensor with the acquisition system. Due to low-bandwidth noise, which is impossible to filter without affecting the measurement signal, the input voltage of the bolometer is modulated to bring the information content in high frequency and consequently increase the signal-to-noise ratio. Then, with a synchronous IQ demodulation the original signal is recovered. However, since the transmission line distortion is not negligible, its effect needs to be taken into account during the demodulation process. In this paper, a possible adaptation of the synchronous IQ demodulation to compensate for the transmission line distortion is presented. Using a square wave as the modulation signal, it is possible to recover the amplitudes of the distorted first and third harmonics, and by knowing the ratio between the first and third harmonics of the square wave signal, it is possible to make a frequency-based online identification of the transmission line (similar frequency-based techniques can be applied also in mechanical systems [2]). This method can identify the transmission line at experiment time, allowing also to cope with possible changes in the frequency response of the transmission line during the pulses due to thermal effects. Therefore, distortion effects get compensated giving a precise estimate of the power radiated by the plasma. The effectiveness of the proposed approach is demonstrated with simulations and finally, a real-time implementation of the whole modulation-demodulation process with transmission line compensation using the envisaged real-time framework for ITER diagnostics and a Wheatstone bridge is shown

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Exploring Runaway Electron-PFC Impacts Using Electron Beam Experiments and Modelling

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Runaway Electrons (REs) are beams of relativistic electrons which can occur during transient events in tokamaks. These beams can carry 10s-100s of MJ of energy in power plant scale fusion reactors, and have caused wall melting, vaporization, and even a superconducting magnet quench in previous tokamaks. REs pose one of the greatest threats to plasma facing components (PFCs) due to their tendency to deposit energy in localised wetted areas and their ability to penetrate far into PFC materials due to individual REs having energies in the MeV range. Understanding what these impacts may look like in future devices is essential to creating mitigation strategies. The HEAT RE module was created to simulate these impacts and their consequences. This module connects runaway electron transport models directly to the design and engineering of plasma facing surfaces using both 2D and 3D analysis. In this work experiments using an electron beam and former Alcator C-Mod experiments are used to validate results from HEAT. Using a benchtop electron beam, electrons in the MeV energy range are shot at Tungsten tiles to explore melt dynamics. Total energy deposited, timescales of deposition, and the energy of electrons in the beam are varied to explore how different RE beam scenarios affect potential melt patterns. In addition, a discharge with an RE beam which caused melting in Alcator C-Mod is simulated in HEAT. Results are compared to documented melting and used to validate the RE module in HEAT.

This work is supported in part by Commonwealth Fusion Systems.

Latest developments in DTT soft X-ray diagnostic

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

The Divertor Tokamak Test Facility (DTT), currently under construction at ENEA in Frascati (Rome, Italy), represents a milestone in the EUROfusion roadmap towards ITER and DEMO. This superconducting tokamak is designed to develop and validate advanced heat and exhaust management strategies.

Diagnostics, such as Soft X-Ray (SXR), are therefore a crucial asset to achieve the scientific objectives of the DTT experiment. Advanced studies have been conducted on CVD diamonds, with tests planned experimental tomography tests expected to provide valuable insights into their optimization. While the conceptual design of the SXR diagnostic is complete, recent thermomechanical simulation have highlighted the potential need for an active cooling system to ensure the functionality of SXR system. This study explores an alternative layout that includes an actively cooled support for the SXR detectors, demonstrating how both thermal performance and optimal use of the available space in the ducts can be achieved. Additionally, the proposed design has been tested for compatibility with other high-priority systems such as remote handling, first wall active cooling, in vessel coils cabling, and baking system.

Independently of this new layout, the literature highlights how SXR data can be used for various purposes, such as retrieving emissivity profiles for specific studies (e.g., MHD instabilities or impurity localization). To achieve these goals, well-established tomographic inversion methods can be employed. While the lines of sight have already been optimized for the conceptual design, further studies are ongoing to explore possible improvements in poloidal coverage.

In this work, significant advancements to the reconstruction software used for SXR data are reported. Notably, an iterative Tikhonov-based inversion method for tomography has been developed, and the pool of regularization matrices available for testing more complex synthetic emissivity distributions (phantoms) has been expanded to scenarios with up to 45 MW of injected external heating. These enhancements have enabled the generation of more accurate and precise tomograms compared to previous implementations based on the same method.

Finally, it is important to emphasize that the studies and results presented in this paper are expected to contribute to the realization of a synthetic SXR diagnostic, complementing the bolometric diagnostic, which is currently under development.

Exploring advanced divertor configurations as power exhaust solutions for fusion pilot plants using edge simulations of SPARC and ARC

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

Edge simulation codes SOLPS-ITER and SOLEDGE2D are being employed to study advanced divertor configurations in the SPARC and ARC contexts, testing power exhaust solutions in Fusion Pilot Plant (FPP) relevant conditions and predicting heat flux densities to the plasma facing components (PFCs). The power exhaust projected for future tokamak-based power plants presents a major challenge for the realisation of fusion energy; FPPs will need high radiation in the Scrape-Off Layer (SOL) to reduce divertor plasma temperatures/fluxes to levels that project to realistic lifetime erosion rates. Advanced magnetic divertor configurations are studied as advantageous potential solutions, including double-nulls, long-legs and magnetic field flaring with secondary divertor X-points. Significant benefits of these geometries have been observed in previous edge modelling for ARC [1] and EU-DEMO [2], and in experiments [3].

SPARC is a compact high-field tokamak under construction by Commonwealth Fusion Systems and partners. SPARC will have a dedicated Advanced Divertor Mission, with a moderately flexible divertor that will allow a variety of magnetic topologies to be explored, and able to achieve divertor metric Psol*Bt/R values up to ~200 MWTm-1 - comparable levels to ARC (>170) and EU-DEMO (~100) designs. SPARC will therefore be a next-step experimental device for testing integrated divertor scenarios under power-plant-relevant SOL conditions.

A modelling workflow has been developed for SPARC and ARC edge simulations to analyse advanced divertors in FPP-relevant contexts. SOL similarity metrics are used to identify ARC-matched SOL conditions in SPARC's boundary plasma operational space. Magnetic equilibria have been generated for standard divertor, X-divertors, and X-point Target (XPT) divertor configurations in SPARC, consistent with its poloidal field coils design, and used to construct SOLPS-ITER simulations. Scans of injected radiating impurity are performed, and the relative capability of these configurations to reduce divertor heat flux densities to acceptable levels under an ARC-relevant exhaust challenge are compared.

The ARC design provides an integrated compact FPP context for evaluating advanced divertors as a power exhaust solution on the power plant scale. ARC's characteristic divertor design is the long-leg double-null XPT, which is found to have the potential to reduce target plasma temperatures by factor ~10 without impurity seeding [1]. The SOLEDGE2D code, with its unique capability to include up to 6 magnetic X-points in the plasma domain, is employed to perform the first full-domain simulations of this advanced divertor concept. Vertical and horizontal orientations for the long outer divertor leg geometry, integrated into the reactor blanket, are considered and exhaust performance benefits are evaluated between these potential design choices. The simulations are used to predict power and particle loadings over the full ARC inner vessel PFCs, including both the divertor volume and main chamber walls, and can identify localised heating/erosion regions of potential concern to help inform future inner vessel designs.

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Hydrogen Retention Characterization of Lithiated Porous Tungsten Samples Using the Sample Exposure Probe in LTX- β

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

Hybrid liquid lithium – porous tungsten materials are being considered as candidate plasma facing components due to their ability to maintain a continuously replenishable low- Z plasma interface while tolerating the high steady-state and transient heat fluxes encountered in a fusion reactor environment [1]. Porous tungsten samples were fabricated by Spark Plasma Sintering (SPS) using tungsten powders measuring between 10 and 15 μm which resulted in substrates approximately 70% dense with pores in the order of tens of micrometers. Understanding the hydrogen retention of these lithiated micro-scale porous tungsten is important for future reactor applications since it affects both plasma performance and tritium safety limits.

Hydrogen retention studies are being conducted using the Sample Exposure Probe (SEP) [2] in LTX-β. The SEP has the capability of performing in situ Temperature Programmed Desorption (TPD) immediately after a sample has been exposed to LTX-β plasmas, and in vacuo surface science characterization such as X-ray Photoelectron Spectroscopy (XPS). The SEP has been upgraded from its previous two iterations to use electron beam heating instead of resistive heating, which improves its TPD's resolution, and to have a more robust design capable of withstanding the conditions in a fusion reactor. Lithiated porous tungsten samples are being exposed on the midplane of the low- field side of LTX- β at solid and liquid states to a fluence of ~ 10^{22} /m². An identical porous tungsten sample with no lithium will also be used as a control. TPD will be used to quantify the amount of hydrogen retained in the samples, comparing the hydrogen retention between solid and liquid lithium states in the porous matrix, as well as comparing to previous SEP results using solid tungsten.

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Impact of geometry and operating parameters of propellant valves on pellet launch and acceleration in shattered pellet injection technology

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Tuesday Posters 1, Lobdell (Building W20 Room 208), June 24, 2025, 10:30 AM - 12:00 PM

Shattered Pellet Injection (SPI) is a common technique, used in Disruption Mitigation Systems to prevent or minimize the effect of plasma disruptions in Tokamaks [1] by accelerating a large cryogenic pellet and shattering it prior to the plasma entrance, creating a plume of smaller fragments. The operation of an SPI system necessitates a fast-acting valve to shear the pellet off the barrel wall and to accelerate it to a desired velocity [2].
Achieving a sufficient pressure peak at the rear of the pellet is crucial for dislodging the pellet, while the pellet velocity depends mainly on the gas amount released by the valve [3]. These parameters are determined by the internal geometry of the components along the propellant gas path leading to the pellet and by the operation parameters [4]. The former includes the internal volume of the valve, the orifice and the valve plug as well as the breech volume. The latter are mainly influenced by the initial valve pressure and the piston opening characteristics.
To gain a deeper understanding of these relationships, we have conducted both Computational Fluid Dynamics (CFD) simulations and laboratory measurements utilizing our eddy current actuated fast valve [5] at its novel dedicated test bed [6] equipped with interchangeable components allowing different geometrical setups. This contribution discusses the outcomes of these simulations and laboratory tests.

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Current status and challenges of the development of plasma-facing components

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Tuesday Parallel 2a - Divertors and Plasma Facing Components I, Kresge Main Theater (Building W16, upstairs), June 24, 2025, 2:00 PM - 3:30 PM

The severe operating conditions for Plasma-Facing Components (PFCs) in future power-generating fusion devices require the development of advanced materials and components. PFCs must not only withstand high steady state power loads of up to 20 MW/m², but also a high number of thermal cycles and shocks. In addition, the design of PFCs and the selection of appropriate armour and structural materials must take into account the change in thermo-mechanical properties due to damage, activation and transmutation by fusion neutrons. At present, water-cooled PFCs are foreseen in most future fusion devices to provide reliable heat removal capability and to allow only moderate extrapolation of the technologies developed and tested for ITER. However, attempts have been made to optimise the design, as well as the armour and heat sink materials, with a view to future applications under even harsher conditions. This contribution gives an overview of the requirements for plasma facing components and the state of the art solutions. In addition, novel experiments on the mechanical properties of W under synergistic loading as well as new concepts and materials developed at the MPI for Plasma Physics will be presented. The latter include investigations on W composites and alloys as well as innovative fabrication methods such as additive manufacturing and cold spray coating of PFC mock-ups, which were finally qualified under relevant power loads in the high heat flux facility GLADIS.

Hydrogenic Isotope Distillation and Accounting in the Actively Pumped Open-Surface Lithium LOop (APOLLO)

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Tuesday Parallel 2a - Divertors and Plasma Facing Components I, Kresge Main Theater (Building W16, upstairs), June 24, 2025, 2:00 PM - 3:30 PM

Liquid lithium as a plasma-facing component (PFC) is attractive due to improved plasma performance, high potential for energy flux absorption and avoidance of failure modes that plague traditional solid PFCs such as melting, integrity degradation, and loss of thermal contact. Lithium PFCs are low-Z, have a high gettering potential for hydrogen and impurities, and allow access to a low-recycling regime. However, lithium's uptake of hydrogenic species can be problematic for future fusion plants due to strict limits on tritium inventory. To address this issue, the University of Illinois at Urbana-Champaign (UIUC) with Tokamak Energy is developing the Actively Pumped Open-Surface Lithium LOop (APOLLO) comprising a flowing lithium loop, a free-surface PFC within a magnetic field, a deuterium plasma source or electron beam heating, and a distillation column for the extraction of hydrogenic species. The current PFC module consists of a computationally optimized distributor that evenly injects lithium into an 8 cm wide 3D-printed ordered mesh inside a free surface flow channel. The PFC module can be exposed to an Electron Cyclotron Resonance (ECR) hydrogen/deuterium plasma source that is characterized with an array of 16 Langmuir probes, a Retarding Field Energy Analyzer (RFEA), and actinometric spectroscopy. Alternatively, the PFC free surface can be exposed to an electron beam capable of supplying a heat flux greater than 10 MW/m2. The plasma-exposed liquid lithium enters a collector at the end of the plate, and flows through a distillation module, which is the focus of this work. The inductively heated Hydrogen Distillation Experiment (HyDE) is a vacuum device in which contaminated lithium is thermally treated at temperatures above 700℃ to remove hydrogenic species and impurities. This presentation reports on ECR plasma and electron beam characterization, hydrogen accounting in steady state operation, verification of TEMHD flow and dryout resistance, and hydrogen removal from liquid lithium for various initial hydrogen concentrations.

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Design of DTT divertor plasma-facing components

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Tuesday Parallel 2a - Divertors and Plasma Facing Components I, Kresge Main Theater (Building W16, upstairs), June 24, 2025, 2:00 PM - 3:30 PM

The Divertor Tokamak Test facility (DTT) [1] is a fusion device under construction in Italy with the support of the EUROfusion consortium, dedicated to testing alternative divertor concepts under integrated physics and technological conditions relevant to DEMO. Several divertors which may differ in design or/and technologies or/and poloidal profile will be tested during the life of the machine. The first divertor aims to identify promising plasma scenarios, and will have to accommodate strike points, located at various positions according to the different equilibria. Thus, the entire divertor plasma-facing surface is made of W monoblocks joined on CuCrZr pipes (plasma-facing units, PFUs) using the ITER-like target design and technologies. Three components can be identified: the inner, the central (or dome) and the outer target, the latter consisting of outer vertical target and an outer horizontal target. One PFU of each component is joined in series and then fixed to the back-plates that have the role of interface with the cassette body. The PFU monoblocks have a minimum thickness of 3 mm on the plasma side. In this way the surface temperature can remain lower allowing to dissipate high heat flux without the recrystallization of the W, reducing the plastic deformation and the vulnerability to cracks.
To avoid local damage of protruding leading edges due to the gaps between the monoblocks and to assembly and manufacturing tolerances, a toroidal monoblock bevel has been implemented. The resulting admissible reduced total load on the monoblock has been evaluated and compared with those predicted by SOLEDGE2D-EIRENE code simulation: even partially detached scenarios can be sustained by the targets in steady state (provided that steady state is compatible with the increased W erosion and associated core contamination) for the three reference magnetic configurations (SN, X-D and NT). This ensures an adequate safety margin for divertor operations even in case of loss of detachment.

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US ITER Update and Value for Fusion

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Tuesday Parallel 2b - Design Integration, Construction, Assembly, Commissioning, and Lessons Learned I; Control, Sala de Puerto Rico (Building W20 Room 202), June 24, 2025, 2:00 PM - 3:30 PM

The United States is delivering essential systems for ITER operations and science, including the superconducting central solenoid magnet for the center of the machine, considered the “heartbeat” of ITER. Four of the six modules required for the central solenoid are already on site and stacked for assembly; the remaining two modules will be delivered later this year. Manufacturing for electron cyclotron heating transmission lines is underway, and deliveries are continuing for vacuum and roughing pumps systems. Following Department of Energy approval in late 2024, additional systems are preparing for fabrication activities. Diagnostics, fueling and tritium processing systems are advancing designs in preparation for manufacturing. Oak Ridge National Laboratory manages US ITER for Fusion Energy Sciences (DOE Office of Science) in partnership with Princeton Plasma Physics Laboratory and Savannah River National Laboratory. Overall, U.S. contributions to the ITER project are ~50% complete.

The ITER project crosscuts the nation’s fusion goals for research, technology development and a path to practical fusion energy. For 9.09% of construction and 13% of operations, the United States receives 100% of ITER science and intellectual discovery. ITER will provide data essential to multiple fusion configurations and is relevant to a broad range of private fusion sector needs. As an ITER member, the U.S. gains industry capability in manufacturing for fusion systems, including the development of national supply chains and first-of-a-kind fusion technologies. The project is also delivering practical experience with assembly and integration of an industrial-scale fusion machine licensed by the French nuclear authority. ITER know-how and intellectual property supports a variety of fusion needs and supports risk reduction for future devices and plants. Once online, ITER will provide a comprehensive fusion R&D resource for U.S. fusion, with flexible operations and extensive diagnostics for diverse research needs. The workforce training and growth that has resulted from US ITER and international project is contributing to multiple fusion sectors and influencing the trajectory of the next generation of fusion leaders.

US ITER progress and the intersection of ITER with national fusion goals support the advancement of practical fusion energy systems. The scientific and engineering knowledge gained from U.S. participation in ITER is contributing now to public-private partnerships and industry needs through information sharing and other opportunities.

Lessons Learned from ITER Construction Phase

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Tuesday Parallel 2b - Design Integration, Construction, Assembly, Commissioning, and Lessons Learned I; Control, Sala de Puerto Rico (Building W20 Room 202), June 24, 2025, 2:00 PM - 3:30 PM

The sector sub-assemblies of the Tokamak machine components in the assembly hall are advancing for transfer to the Tokamak pit in order to proceed with the Tokamak machine assembly. The installation of the plant systems are progressing in the Tokamak Complex buildings outside of the Tokamak pit and in the auxiliary buildings towards the Start of Research Operations.

The Lessons learned from construction phase are not limited to construction execution only as they comprise also the return of experience such as from the systems design phase with its maturity, procurement strategy, contract management as well as design requirement.

In order to allow a prompt decision making during contract execution respecting cost and schedule, empowered staff to be assigned who represent with one voice the Organization in front of the contractors to allow prompt decision making in the construction phase.

The paper summarizes the Lessons learned which have been gained during the construction phase and provides its feedback from the systems design phase with its requirements, contract preparation and execution.

TORAX: A Fast and Differentiable Tokamak Transport Simulator in JAX

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Tuesday Parallel 2b - Design Integration, Construction, Assembly, Commissioning, and Lessons Learned I; Control, Sala de Puerto Rico (Building W20 Room 202), June 24, 2025, 2:00 PM - 3:30 PM

We introduce TORAX, an open-source differentiable tokamak core transport simulator targeting fast and accurate core-transport simulation for pulse planning and optimization, and unlocking broad capabilities for controller design and advanced surrogate physics. TORAX is written in Python using JAX, and solves coupled time-dependent 1D PDEs for core ion and electron heat transport, particle transport, and current diffusion. JAX's just-in-time compilation provides fast computation, while maintaining Python's ease of use and extensibility. JAX auto-differentiability enables gradient-based optimization techniques and trajectory sensitivity analysis for controller design, without time-consuming manual Jacobian calculations. JAX's inherent support for neural network development and inference facilitates coupling ML-surrogates of physics models, key for fast and accurate simulation. Code verification is obtained by comparison with the established RAPTOR code on ITER-like and SPARC scenarios. Application of TORAX is demonstrated within a pulse planning workflow designed for optimization of SPARC actuator trajectories. TORAX is an open source tool, and aims to be a foundational component of wider workflows built by the wider community for future tokamak integrated simulations.

Modelling an in-cryostat LOCA from a superconducting magnet

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Tuesday Parallel 2c - Magnets and Cryogenic Systems II, Kresge Little Theater (Building W16, downstairs), June 24, 2025, 2:00 PM - 3:30 PM

Accidental transients in the superconducting magnet systems of nuclear fusion reactors require thorough analysis, as they can jeopardize the integrity of critical components. These components, such as toroidal field coils, are among the most expensive in the whole plant, and often impossible to repair. Among these scenarios, the in-cryostat Loss-Of-Coolant Accident (LOCA) is particularly critical. This event, triggered by the release of pressurized supercritical helium (SHe), has been observed in experimental facilities such as ITER's Central Solenoid Module (CSM) testing and during the JT-60SA tokamak commissioning. During such a LOCA, the coolant is discharged into the cryostat, which is maintained under vacuum, forming an underexpanded supersonic helium jet. Due to the thermodynamic properties of helium, this rapid expansion results in a multiphase jet comprising supercritical, vapor, and liquid phases. Assessing potential damage to the magnets and developing diagnostic capabilities for such incidents requires detailed knowledge of the 3D evolution of pressure and temperature distributions. Consequently, a 3D CFD model must be developed to capture the complex spatial and temporal dynamics of both cryostat pressurization and jet propagation.
To address these challenges, a 3D transient analysis of the in-cryostat LOCA is carried out, based on a simplified ITER CSM geometry. The model captures the development and propagation of the SHe jet, starting from the moment of rupture and continuing for a few minutes, allowing sufficient time for the module to heat up. The physical models, particularly the multiphase model, are validated against a 2D benchmark problem before being applied to the more complex 3D scenario. Additionally, an Adaptive Mesh Refinement (AMR) algorithm is employed to dynamically track the expected shock fronts.
Particular attention is given to the temperature distribution in the CSM, with its evolution monitored from the onset of the accident to the end of the transient.

An overview of the STEP remountable joints technology program

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Tuesday Parallel 2c - Magnets and Cryogenic Systems II, Kresge Little Theater (Building W16, downstairs), June 24, 2025, 2:00 PM - 3:30 PM

The STEP powerplant has downselected a vertical maintenance strategy that relies on the use of remountable joints to segment the TF coil for access to the machine. The radial build of the reactor is sensitive to the performance of joints, cables and the methods used to clamp the joints. Rapidly de-risking these design elements has become a corner stone of the program's focus. A technology development program has been established to address the challenges associated with the relatively low TRL of remountable joints operating at high currents and high fields and to feedback learnings to the concept design.

Some of the projects under this program are discussed briefly and their progress is outlined. Joints have been prototyped between low and high current cables; this forms the connection prototyping project. Multi-turn, multi-pancake segmented coils (Baby Toro) have been developed under the array prototyping project to improve model reliability. A swath of small tests has been conducted to build up the understanding of edge-stacked tape joints and the operational resilience of the remounting surface under the physics and operations project. Manufacturing trials, proof of principle and operation have been conducted under the clamping project. Early progress includes the development of a sub-scale remountable joint at 5kA and VPI manufactured cable with a remountable joint tested at SULTAN. Early multiphysics models have been built to estimate joint resistance and current distribution.

Long term plans have been mapped to realise remountable joint magnetic technology to enable STEP's the vertical maintenance strategy. Major milestones between now and the commissioning of STEP in 2040 include de-risking joint resistance and remote operability on sub-scale STEP TF coil (TFMC) and later, a full-scale TF coil prototype. Alongside the TFMC program, there are focused programs to understand the load-handling capability of the joints, surface repair methods and remote maintenance trials.

Development and operation of a 5.6 T ReBCO-based levitated dipole magnet system with HTS flux pump excitation

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Tuesday Parallel 2c - Magnets and Cryogenic Systems II, Kresge Little Theater (Building W16, downstairs), June 24, 2025, 2:00 PM - 3:30 PM

Levitated dipole configurations offer a distinct approach to magnetic fusion, promising high-beta plasma confinement and natural stability properties. This work presents the development of a High-Temperature Superconducting (HTS) levitated dipole magnet system designed to advance fusion energy research beyond the achievements of the LDX and RT-1 programs. Our novel implementation utilizes ReBCO technology and HTS flux pumps to create a 5.6 T DC dipole field, matching the confinement capabilities of traditional LTS systems, with the system designed for sustained levitation operation of more than 2 hours.

The experimental program demonstrates progressive validation of dipole confinement physics through two key campaigns. Initial tests achieved stable plasma confinement for over 20 seconds using a mechanically supported dipole magnet operating at 40% rated current. The second campaign addresses the fundamental challenge of dipole fusion: simultaneously maintaining plasma confinement while levitating a 500 kg magnet at approximately 1 m above the chamber base with 60% rated current. The experimental measurements of magnetic field, operating current, magnet/coil voltages, and magnet temperatures enabled us to evaluate the joint resistances, thus defining the voltage required to reach the full field operation. These results provide detailed characterisation of the magnet system's performance, including zero-field region operation within the magnet, levitation stability, and the unique behaviour of non-insulated HTS coils under flux pump excitation.

This research establishes critical engineering benchmarks for scaling levitated dipole fusion devices, demonstrating that HTS technology can effectively address the complex requirements of stable magnetic levitation while maintaining the field geometry necessary for dipole confinement. The validated performance metrics and operational insights advance the development pathway for dipole-based fusion energy systems.

Laser-patterned magnets for high-precision stellarator fields

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Tuesday Parallel 2c - Magnets and Cryogenic Systems II, Kresge Little Theater (Building W16, downstairs), June 24, 2025, 2:00 PM - 3:30 PM

Stellarators demand complex magnets for generating three-dimensional fields for plasma confinement, presenting major manufacturing challenges. Renaissance Fusion addresses these by developing cost-effective laser-engraved wide conductors wound on cylindrical modules, shaping current distributions to accurately produce the desired magnetic fields. Aluminum and copper prototypes validated the engraving pattern optimization tools before their extension towards High Temperature Superconductors (HTS) applications. The first prototype demonstrated the concept by reproducing a gyrotron field with a mean local error of 10^{-2} using stacked aluminum rings [1]. In the second experiment, laser-patterned copper coils achieved a uniform MRI field with 10^{-5} precision in an 8 cm spherical region, verified by means of an NMR probe [1]. The current third prototype focuses on replicating the magnetic field of the Wendelstein 7-X (W7-X) stellarator, a benchmark for complex non-axisymmetric configurations. The prototype includes five engraved cylindrical coils and auxiliary planar coils surrounding a 1.5 m toroidal vacuum vessel, targeting 0.1 T in the magnetic axis. The thermal behavior of the coils was modeled to ensure a temperature rise below 100 K despite 15kW dissipation, avoiding critical structural stresses. Additionally, studies showed that the laser-patterned current leads contribute a maximum local field error of 400 ppm, staying within the 1000 ppm target. The vessel ensures high-vacuum compatibility for Electron Beam Mapping (EBM), a technique used in W7-X [2] and MUSE stellarators. EBM visually assesses magnetic fields by tracing nested flux surfaces through electron collisions with a phosphorescent surface, enabling the comparison between produced fields and target configurations. The design, construction, and initial experimental results are presented. This project highlights the potential of laser-patterned magnets to streamline stellarator coil manufacturing and reduce costs. Future work will replicate the process with laser-engraved wide HTS tapes, currently being developed at Renaissance Fusion, improving magnetic field performance and fusion power plant feasibility.

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Expected performance of the JT-60SA edge Thomson scattering diagnostic in OP2

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Tuesday Posters 2, Lobdell (Building W20 Room 208), June 24, 2025, 2:00 PM - 3:30 PM

The Edge Thomson Scattering (ETS) diagnostic system for JT-60SA is designed to provide precise measurements of electron temperature (Te) and density (ne) profiles in the plasma's pedestal and edge regions, which are critical for understanding confinement and stability in fusion plasmas. The system is optimized to achieve a spatial resolution of 5 mm in steep gradient regions and measure Te and ne with precisions better than 10% and 5%, respectively, for Te ranging from 10 eV to 10 keV and ne of 1019 m-3. As JT-60SA approaches its operational phase 2 (OP2), predicting the ETS system's performance under realistic plasma conditions is crucial for optimizing its diagnostic capabilities.

This work presents a synthetic diagnostic developed to evaluate and predict the performance of the ETS system during OP2. The approach models the entire system, including optical and detection elements, while accounting for plasma light background and noise. Key performance metrics, such as signal-to-noise ratio and measurement errors in Te and ne, are analyzed through these simulations.

The results provide a detailed understanding of the expected performance of the ETS system under various operational scenarios. The insights gained supports the system's readiness for OP2. Additionally, they offer valuable feedback for commissioning, calibration, and efficient utilization of the ETS diagnostic.

High-Power Helicon System Upgrades and Repairs at DIII-D

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Tuesday Posters 2, Lobdell (Building W20 Room 208), June 24, 2025, 2:00 PM - 3:30 PM

Helicon plasma waves offer a promising method for plasma heating and current drive in fusion reactors. The DIII-D tokamak is equipped with a helicon system operating at 476 MHz, capable of delivering up to 1.2 MW of power. During FY23 operations, several technical challenges arose, including damage to in-vessel thermocouples and RF probes near the antenna. Damaged thermocouples were removed, and the remaining ones were rerouted behind tiles for additional protection. All damaged RF probes were replaced and recalibrated during the FY24 vent. Additionally, issues were identified on the 150° vacuum side of the transmission lines, where loose screws on the inner conductor caused contact with the septum, rendering the 150° port inoperable. The team removed a bent bellows from the inner conductor and replaced the damaged conductor at the 150° port. A non-pressurized waveguide switch was also removed, and a pressurized patch panel was installed to restore the helicon system's functionality, enabling power feed from both the 150° and 210° directions.

Overheating incidents were observed in the elbow of the 210° port during prolonged high-power shots in FY23, though no visible damage occurred. To mitigate this, both 150ºand 210º elbows were copper-plated and diamond-coated to prevent multipactor effects.

In FY24-25, the team successfully demonstrated electron heating through ECE channels in both L- and H-mode, achieving 0.8 keV on-axis heating from the helicon system. Additionally, the team demonstrated the helicon system's ability to couple up to 700 kW of power to the plasma from the 150° feed side for 2 seconds.

The repairs and advancements have fully restored the helicon system's operational capacity, significantly enhancing its potential for plasma heating and current drive in future fusion experiments.

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The effect of boronization on tritium retention in SPARC

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Tuesday Posters 2, Lobdell (Building W20 Room 208), June 24, 2025, 2:00 PM - 3:30 PM

SPARC is a deuterium-tritium (D-T) tokamak designed to produce net fusion power and develop a pathway for commercially viable fusion energy. Designed with a tungsten first wall, the SPARC tokamak is engineered to support vacuum vessel conditioning to achieve high performance plasmas. One conditioning capability deployed on SPARC is boronization, where deutarated diborane will be used to deposit a layer of boron on the first wall.

While boronizations have been conducted on D-D fusion devices such as Alcator C-Mod [1], MAST [2], ASDEX [3], DIII-D [4], WEST [5], EAST and HT-7 [6] this will be the first time boronization is utilized on a D-T device. Ahead of SPARC campaigns, CFS is collaborating on studies to understand hydrogenic inventory management. Two investigations are currently underway to enable prediction of the boronization effect on tritium retention in SPARC; a CFS-UKAEA project and a CFS-PPPL INFUSE project.

The studies presented here are being conducted using the UKAEA's DELPHI facility located at the UKAEA Culham Campus. DELPHI exposes samples to a beam of hydrogen isotope ions which is extracted from a plasma, allowing control over the ion energy and flux. Samples of boron coated tungsten have been produced by plasma deposition and are exposed on a temperature controlled stage. Exposed samples are analysed by Thermal Deposition Spectrometry (TDS) or by Liquid Scintillation Counting (LSC) following sample oxidation in the Tritium Analysis Laboratory (TAL). The initial DELPHI results using boron coated tungsten samples are presented and compared to results from the INFUSE project which uses boron powder samples exposed to either deuterium gas or ions.

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Reliability Optimised Blanket Using Simulation & Test: A Novel Approach to Breeder Blanket Design

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Tuesday Posters 2, Lobdell (Building W20 Room 208), June 24, 2025, 2:00 PM - 3:30 PM

Many public and privately funded fusion reactors are due to begin operation around the middle of the 21st century, the vast majority of which will rely on a deuterium and tritium fuel cycle, with a lithium-containing breeder blanket responsible for providing a sustainable tritium supply. The tritium consumption of large (DEMO-scale) devices will be in the order of 100kg per full power year, several times the current global civil tritium inventory of approximately 35kg. Whilst tritium breeding in fission reactors has supplied experimental devices (the Joint European Torus tritium inventory was limited to 90g), supporting a global fusion industry in this way is not a viable solution. Breeder blanket technologies are immature and are yet to be tested in operational tokamak environments, where tritium breeding and extraction requirements must be met whilst maintaining structural integrity under thermal and electromagnetic loads, plasma disruption events and high neutron fluences. Therefore, it is crucial that blanket design is approached using a methodology with a proven record for success, that minimises risk and ensures thorough exploration of the associated design space.

The aim of this work is to develop blanket concept designs by utilising industry-standard systems engineering processes and methodologies. These include requirements capture, requirements validation and verification, Model-Based Systems Engineering (MBSE), Failure Modes & Effects Analysis (FMEA) and concept generation and selection methodologies. A requirements verification plan lays out a detailed description of the verification activities that need to be executed in order to verify breeder blankets designs – including simple 1D analysis, complex transient 3D high fidelity simulations and a range of experiments and tests. Additionally, an analysis workflow has been developed which links an integrated set of analysis models, which provide low-fidelity concept design verification through a pre-conceptual multiphysics systems simulation, covering neutronics, thermal hydraulics, structural analysis and fuel cycle assessments. Finally, the project has developed a methodology to provide uncertainty quantification for the key performance parameters such as tritium breeding ratio. The output is a range of down-selected blanket concepts corresponding to minimal risk and highest likelihood of meeting the system (i.e. Breeder Blanket) requirements. This has allowed for the exploration of novel areas of design space, alongside the rationale which has led to the decisions.

High Heat Load Retarding Field Energy Analyzer in DIII-D Divertor

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Tuesday Posters 2, Lobdell (Building W20 Room 208), June 24, 2025, 2:00 PM - 3:30 PM

The retarding field energy analyzer (RFEA) probe is used for ion temperature and energy distribution measurements [1]. Its implementation in tokamaks is challenging due to the extreme plasma conditions. At the DIII-D divertor target, the parallel heat flux q// is expected to reach approximately 100 MW/m^2 with electron density (ne) ~ 10^{19} $\#/m^3$ and ion temperature (Ti) ranging from 10 to 200 eV. A new prototype RFEA utilizing the Divertor Material Evaluation System (DiMES) has been designed and constructed for use in these extreme conditions. The design process is guided by thermal finite element analysis (FEA) to enhance the probe's survivability and particle-in-cell (PIC) analysis to improve data accuracy, ensuring optimized overall performance.
A compact geometry is adopted to reduce the probe's total exposure to heat flux by protecting most of the probe body behind the divertor graphite tiles and a tungsten guard plate at the entrance. Due to the concern for heat load, erosion, and material deformation, the entry plate is made with tungsten-rhenium alloy (90%W-10%Re), which maintains superior thermal performance at high temperature [2] while being much less brittle than pure tungsten, and a unique multi-slit entrance opening is used to improve the total ion intake and minimize selective transmission. At the DIII-D divertor target, the typical Debye length (λD) is ~10 µm in the attached divertor condition, therefore it is required that the width of the entrance openings to be around 20 µm, which cannot be produced with conventional manufacturing. With a laser micro-machining technique, five such slits, which are funnel-shaped to reduce particle selective transmission and overall ion loss to the entrance wall, are precisely machined on the 1 mm thick W-Re sheet. To further improve the probe's thermal performance, graphene foils are added in between the guard plate, entry plate, first grid, and insulators to increase heat spreading, while reducing heat transmission to the inner components.
The temperature evolution during typical discharges is simulated by transient FEA (ANSYS) with realistic heat load mapping calculated by the Heat flux Engineering Analysis Toolkit (HEAT), and then validated with infrared (IR) cameras during experiments. Comprehensive PIC simulations are also performed to evaluate the signal distortion caused by selective transmission and space charge accumulation. These distorting effects can be optimized but ultimately unavoidable due to the compactness of the divertor RFEA. It is shown that the I-V signal, under realistic divertor conditions, is likely to diverge from the conventional 1-D interpretation model, which tends to over predict Ti. Thus, a novel PIC-based interpretation method is developed to more accurately reflect the ion energy distribution function [3]. The prototype RFEA is scheduled for testing during the plasma startup phase at DIII-D in 2025, with preliminary results expected to be presented in this session.

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Revised RF chain for density profile plasma reflectometry in SPARC

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Tuesday Posters 2, Lobdell (Building W20 Room 208), June 24, 2025, 2:00 PM - 3:30 PM

The edge-scanning reflectometer [1] is a diagnostic planned for the upcoming tokamak SPARC [2]. This diagnostic, based on the continuous-wave frequency-modulated principle, will measure the density profile at the outboard midplane using a combination of O-mode and X-mode propagation. This system will cover a broad frequency range spanning several microwave bands and fully utilizing each band. The radio frequency (RF) chain consists of a voltage-controlled oscillator, frequency conversion with image rejection for heterodyne reception, and an active delay line equalized to compensate the coaxial cable's attenuation slope, among other elements common to state-of-the-art reflectometer systems. Here, we present the revised RF chain to produce and receive the signals used in the measurement. It has matured since first presented to the scientific community and is now close to finalized taking full advantage of commercial and standard RF connectorized components and integrated circuits to achieve an affordable, robust, and compact design.

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Integration of real-time plasma disruption prediction system at ADITYA-U

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Tuesday Posters 2, Lobdell (Building W20 Room 208), June 24, 2025, 2:00 PM - 3:30 PM

Sudden plasma disruption can damage plasma-facing components in the tokamak vessel which needs to be accurately predicted and measured to avoid damage. An LSTM-based auto-encoder model is utilized for precursor prediction with sufficient warning time at ADITYA-U. It is essential to implement real-time embedded hardware for the integration. This involves real-time triggering of plasma discharge, periodic data logging, recurrent operation-based prediction, and providing output warning for mitigation or control. Nine important diagnostic signals are interfaced as ADC channels, sampled at 1 kHz in a closed-loop system. The model's output inference executes within 500 microseconds, ensuring the entire lifecycle runs at a 1 kHz rate recursively.

The LSTM-based multivariate model is trained offline using approximately 8,000 past plasma discharges. Model inference is performed using object code conversion from the Python/Keras model via the keras2c utility. The MIC-1816 embedded controller, a low-latency input/output and high-computational single-board controller (SBC), is employed to infer the model output in real-time. Disruptive or non-disruptive shots are categorized using a unique classification method. Following classification, time series regression is derived to obtain the anticipation time of the disruption event based on a dynamic threshold. The system is integrated at ADITYA-U tokamak. The deployment is successfully validated with more than 600 real-time plasma discharges. The results indicate warning alarm times within the range of 3 milliseconds to 25 milliseconds for typical plasma discharges lasting up to ~400ms. The design and development process, along with the results, are discussed and demonstrated.

Pre-experimental modeling of corrosion and precipitation processes in a PbLi corrosion loop at ORNL

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Tuesday Posters 2, Lobdell (Building W20 Room 208), June 24, 2025, 2:00 PM - 3:30 PM

Lead-lithium (PbLi) is a leading candidate for liquid breeder blanket concepts in fusion reactors. However, corrosion of Reduced Activation Ferritic Martensitic (RAFM) steel (which is presently considered as the main structural material candidate of blanket) and other candidate steels in high-temperature flowing PbLi under magnetohydrodynamic (MHD) conditions, continues to pose significant challenges in designing robust blanket systems. In response, a PbLi corrosion loop is under development at Oak Ridge National Laboratory, including pre-experimental modeling for future corrosion/precipitation experiments. To support this effort, new computational models in COMSOL are being developed and applied to the analysis of mass transfer processes to guide the forthcoming experimental studies.

This paper presents two modeling studies: (1) Corrosion modeling investigates MHD-induced corrosion of RAFM in PbLi in the rectangular test-section duct. It innovatively couples MHD, heat transfer, and mass transfer to predict material loss from the steel wall and transport of corrosion products in PbLi. Such study enables evaluation of wall thinning, which is crucial to maintaining structural integrity of PbLi channels; (2) Precipitation modeling examines iron deposition on heat exchanger surfaces by integrating fluid flow, heat transfer, and mass transfer to quantify and mitigate potential blockages due to deposition of corrosion products in the cold section of the PbLi loop, which could compromise system performance. Simulation outcomes will be compared with phenomenological models, providing comprehensive insights into fundamental PbLi corrosion and precipitation behaviors.

This is the first successful demonstration of fully coupled PbLi flow, heat transfer, and mass transfer modeling in COMSOL for PbLi corrosion and precipitation studies. In the future, the modeling tools will aid in designing and analyzing full-scale blanket systems.

Mechanical and Thermal Design of ITER's Motional Stark Effect (MSE) First Mirror with RF Cleaning Capability

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To enable the Motional Stark Effect (MSE) diagnostic to measure the magnetic field inside ITER's plasma, the light generated by the interaction of the neutral beam with the plasma must first be redirected out of the vacuum vessel. This requires a plasma-facing mirror capable of withstanding significant inertial, thermal, and nuclear loads while accurately reflecting light to a sequence of downstream mirrors in less exposed environments. Given the challenging conditions and limited accessibility for repair or replacement, a remote cleaning mechanism is essential to remove debris or contaminants from the mirror surface during operation. The cleaning mechanism imposes additional requirements: the mirror must remain electrically isolated, minimize parasitic capacitance to ensure effective RF-based cleaning, and maintain sufficient cooling to limit deformation. Tungsten was chosen as the mirror material for its reliability in cleaning, but this has increased the expected neutronic heating by an estimated factor of twelve. To address these challenges, the design incorporates an aluminum nitride plate to transfer heat efficiently to a copper alloy, water-cooled heat exchanger while maintaining electrical and capacitive isolation. This approach minimizes mirror deformation, keeping it within acceptable limits even under worst-case temperature conditions of 350°C. To maintain conductive cooling across the mirror, consistent pressure across the mirror face is achieved through 28 bolts, each equipped with a carefully engineered fastener stack, ensuring robust mechanical stability and thermal performance. The design under review for MSE's Preliminary Design Review balances stringent optical performance requirements with the extreme thermal and mechanical demands of the environment. Furthermore, it enables the development of a prototype to test and validate critical aspects of the system, including thermal management, structural stability, optical performance, and the functionality of the RF-based cleaning mechanism under simulated operational conditions.

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Tearing instability prediction for MAST shots using a Transformer based deep learning model

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This paper introduces a deep learning model which uses Transformer based architectural components for next time step prediction of plasma state parameters and tearing instability probability. The developed model features modern activation functions and utilises the self attention mechanism to capture long-range spatial dependencies within multi-variate diagnostic signals. The model provides real time uncertainty estimates by predicting probabilistic output distributions via the use of a negative-log-likelihood loss function. Training and evaluation of the model was conducted using shot data from the Mega Ampere Spherical Tokamak (MAST) experimental program. The model offers the capability for the development of uncertainty aware real-time plasma control for instability avoidance.

The effect of high-power transients and plasma sheaths on tungsten coatings used for radio frequency launcher applications*

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Radio frequency (RF) launchers will often use a material coating on their structural components, such as a high-conductivity material on the inside surfaces of the launcher or a high-temperature material for plasma-facing parts of the launcher. The coatings need to be robust enough to survive enhanced erosion from the local plasma sheath or an arc or other transient event from the plasma (e.g., an edge localized mode) without causing a catastrophic failure of the coating. High-power transient effects are being explored by using an RF-induced vacuum arc to determine the robustness of material coatings made by a variety of manufacturing techniques. A 1/4-wavelength resonant section of vacuum transmission line terminated with an open circuit electrode structure with a well-defined electric field (30-40 kV/m) produces repeatable arcing conditions. The transient time of the arc is on the order of microseconds, and the light emitted by the arc has been characterized by a filterscope. The initial focus is on tungsten as a plasma-facing material, including sintered tungsten, additively manufactured tungsten, and tungsten coatings on steel produced via physical vapor deposition (PVD) and functionally graded tungsten/steel coatings deposited by low-pressure plasma-spraying (LPPS). Arcing often initiates on sharp microstructures and causes localized melting of tungsten at the surface of all the materials (sintered, additively manufactured, PVD, etc.) and results in resolidified melt pools with surface cracks. PVD coatings without an interface layer with the steel fail catastrophically from an arc and result in severe delamination of the coating. The effects of the RF plasma sheath on the coating are being determined by using the RF Plasma Interaction Experiment (RF PIE). RF PIE is an Electron Cyclotron Resonance microwave-based plasma source (2.45 GHz, <5 kW) with an RF-biased electrode. Previous results have shown that RF sheaths enhance sputtering of tungsten compared to mono-energetic sheaths, and current work is focused on determining hydrogen retention of sintered tungsten as a function of bias conditions, which has shown an increase in retention with ion energy. Similar experiments are planned for additively manufactured and functionally graded plasma sprayed tungsten and will be discussed.

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Outcome of the Preliminary Design Review for the ITER Radial Gamma Ray Spectrometer

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The ITER Radial Gamma Ray Spectrometer (RGRS) is an ITER diagnostic undergoing its Preliminary Design Review with an
expected deployment for Phase DT1 (2041). RGRS is expected to measure the density profile and energy distribution of alpha-particles through reactions with plasma impurities, the current and maximum energy of runaway electrons through bremsstrahlung emissions, and fusion power via a radiative channel of the DT fusion reaction. The diagnostic employs LaBr3 scintillators coupled with photomultiplier tubes along four radial lines of sight. LiH attenuators are adopted to reduce the background due to direct neutrons.

Performance assessments indicate that RGRS can fulfill its functions regarding runaway electrons, while the feasibility of measuring alpha-particles is uncertain due to the intense gamma-ray background observed at JET. Further study is needed to confirm this measurement possibility. Additionally, while fusion power measurements appear possible, satisfying ITER requirements necessitates detailed knowledge of the gamma-ray-to-neutron branching-ratio of the DT fusion reaction, which will be investigated through dedicated experiments, and possibly gamma-ray attenuators to reduce background.

Prediction of Line Emission from Thailand Tokamak 1 (TT-1) Plasma Using Computer Vision

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Since TT-1 has been tested and operated, no qualitative characterization of impurity radiation has been performed due to lack of the related diagnostics. Only an residual gas analyzer (RGA) and a visible digital camera have been used for overall monitoring of the remaining elements/molecules and visible emission, respectively. Recently, the development of impurity monitoring diagnostics, namely optical emission spectroscopic (OES) diagnostics and absolute extreme ultraviolet (AXUV) photodiode detectors, for Thailand Tokamak 1 (TT-1) have been carried. However, they have yet to be installed at and fully operated on the TT-1. From this, to make use of the recorded digital images of visible TT-1 plasmas for impurity analysis, the color analysis of those images have been conducted under the help of computer vision. In general, a JPEG image consists of the colors described by the RGB color mode, as in connection with the related wavelength via the CIE RGB color space. These captured colors can be correlated to the remaining elements/molecules, i.e. both fuels and impurities. The quantities of impurities of various charge states correspond to the intensity of line emission with respect to de-excitation and neutral recombination. As a result of the standard RGB-to-HSV transformation, the hue (H) should connect with the emitted wavelength, and the saturation (S) and the value (V) should connect to the emitted intensity. The plan is to analyze the color of the emitted region in the digital images of the TT-1 plasma to characterize both fuel and impurities in terms of species and their quantities, and relate them with the help of the recorded RGA data. The presentation of this study in the SOFE 2025 will outline the methodology of such characterization, and report the result from the study in terms of the spectra observed in and the comparison with the recorded RGA data of the TT-1.

However, the established model to match the color of the emitted region in the digital images of a visible plasma has been carried out in the separated project, which collects the standard line emission from several spectral tubes of various single atomic gases using a digital camera and a spectrometer, perform the standard RGB-to-HSV transformation, and subsequently analyze them using the principle component analysis (PCA) with numerical methods.

The application of computational neutronics for the systematic and objective down-selection of breeder blanket concepts in support of the Infinity Two Fusion Pilot Plant physics basis

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The body of academic literature contains a wide range of designs with various tritium breeder materials and their associated coolant systems proposed for a commercial fusion plant based on magnetic plasma confinement. Most of these designs were proposed in literature for tokamak systems. In order to evaluate and compare the feasibility of these proposed designs in the challenging geometric context imposed by the stellarator plasma-magnet system, CAD-based geometry tools were developed for automating the generation of blanket structures given proposed stellarator plasma geometries constrained by the associated magnets. The blanket structure and magnet geometry definitions were used to perform neutronic simulations to assess tritium breeding ratio and marginal neutron shielding performance with relevant plasma modeled as a volumetric DT neutron source. These neutronics performance metrics drive both fuel self-sufficiency and commercial viability. With this methodology, Type One Energy presents a neutronics-driven comparative evaluation of leading breeder blanket designs under stellarator geometry constraints which ultimately lead to down-selection in support of the physics basis of the Infinity Two Fusion Pilot Plant.

Parametric Multiphysics Workflow for Evaluating Tritium Breeder Blanket Performance

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The efficient and accurate design of tritium breeder blankets is a cornerstone for the success of nuclear fusion reactors, playing a vital role in achieving optimal performance and safety. This study introduces a comprehensive workflow that integrates parametric geometry generation, meshing, and multiphysics simulation to analyze tritium breeder blanket architectures. Specifically, the workflow is applied to evaluate three prominent designs: the Helium-Cooled Pebble Bed (HCPB), Helium-Cooled Lithium Lead (HCLL), and Dual-Coolant Lead–Lithium (DCLL) concepts, with an emphasis on sensitivity analysis.

The proposed workflow begins with parametric geometry modeling using Parablank, a tool developed by IDOM for UK Atomic Energy Authority (UKAEA), which enables flexible design explaoration to analyse to specific requirements. These geometries are subsequently meshed using Salome (open-source meshing software), ensuring the generation of high-quality meshs suitable for simulations. The Fusion ENergy Integrated multiphysi-X (FENIX) code, developed by Idaho National Laboratory, serves as the core simulation framework, integrating key physics domains: neutronics (via OpenMC), thermal-hydraulics, and thermomechanics. This integrated approach facilitates a comprehensive evaluation of breeder blanket performance under fusion re-actor conditions.

The analysis focuses on critical performance metrics, including neutronic heat deposition, Tritium Breeding Ratio (TBR), Displacement Per Atom (DPA), thermal performance, structural integrity, and coolant behavior within the blanket modules. The derived critical metrics enabled comprehensive correlation between scaled mock-ups and full-scale breeder blankets, providing insights for experimental stages and how impacts at fusion reactor level.

Operation of ITER prototype pressure gauges at exceptional pressure increases

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Hot cathode ionization gauges will measure the neutral gas pressure in the vacuum vessel of ITER. Overall, 52 gauge heads based on the concept of the ASDEX pressure gauge but using novel ZrC emitters are located in the divertor, in equatorial ports and in pumping ducts. The high number of gauges installed considers the limited lifetime of the gauges of 860 h and the corresponding need for a high redundancy to meet the requirements on availability of the diagnostic during the whole life time of ITER. The concern has been raised that exceptional pressure increases during events like mitigated disruptions may damage the gauges.

Disruption mitigation will be an active research topic on ITER to optimize the methods for reducing thermal and mechanical loads on the vacuum vessel as well as on first wall components. The method of choice is the injection of shattered pellets which drastically increases the plasma radiation and leads to a fast termination of the discharge without localized thermal loads. The shattered pellets also result in a significant increase of the neutral particle density in the plasma boundary layer where the pressure gauges are operated. The pressure increase has been estimated to reach 100 Pa within 10 ms and further increases to 230 Pa within 5 to 10 s. As these pressures are significantly higher than the gauges' measurement range of 20 Pa, dedicated tests have been performed to check the behaviour of a prototype gauge at such high pressures, to test how fast it can be shut off and if any damages would occur.

The ITER prototype gauges based on a ZrC emitter were exposed to pressure pulses to reach first 100 Pa and subsequently 230 Pa. No detrimental effects could be observed, neither when exposing the gauge to the pressure pulses in an off-state nor when operating it for a short time (order of 10 s) under such exceptional conditions. The shut-off time of the prototype was in the order of 1 s, limited by the parameters of the power supplies used. Accordingly, it can be assumed that with the implementation of a fast shut-down based on the disruption mitigation trigger the gauges in ITER can safely stop their operation before pressures above 100 Pa are reached and will be able to re-establish operation after the disruption mitigation system is triggered.

Spectroscopic Characterization of Plasmoid Properties During Pellet Fuelling in W7-X

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Pellet injection is a widely used technique for core fueling in magnetic confinement fusion devices, such as stellarators. In particular, the Wendelstein 7-X (W7-X) stellarator aims to achieve steady state operation which includes optimized particle fueling with the use of cryogenic hydrogen pellets. Understanding the dynamics of pellet ablation and the resulting plasmoid characteristics is crucial for optimizing fueling strategies and improving plasma confinement. This work presents the spectroscopic characterization of pellet ablation in W7-X, focusing on the Balmer series emissions from hydrogen.
A spectroscopic diagnostic system was employed to monitor the ablation process during the 2024 operational campaign (O.P. 2.2). The system was set to capture Balmer series lines from the hydrogen pellets. These emission lines provide valuable insights into the temperature, density, and velocity of the plasmoid formed during pellet ablation. The analysis of the line profiles allows us to extract key parameters of the plasmoid, such as its temperature, density, and drift velocity.
The insights gained from this research are essential for improving the accurate description of the dynamics of the fuel particles during pellet injection.

Ion temperature measurements in the Scrape-Off Layer using gas-puff based Charge Exchange Recombination Spectroscopy systems at the ASDEX Upgrade tokamak

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Accurate measurement of ion temperature (Ti) in the Scrape-Off Layer (SOL) is a long-standing challenge in fusion research. It has significant implications for turbulence studies in the SOL, quantification of heat fluxes and sputtering effects on plasma-facing components and determination of boundary conditions for integrated models. This work addresses the challenge by developing a novel diagnostic technique focused on the separatrix region, based on Charge Exchange Recombination Spectroscopy (CXRS) [1]. A main ion (here, D) gas-puff CXRS system, which uses injection of room-temperature neutrals, offers enhanced signal-to-noise ratios in the near separatrix and SOL regions.

The use of different donor neutral species (D or He) and the observation of different visible line transitions, here Dα ($n = 3 \rightarrow 2$) and Dγ ($n = 5 \rightarrow 2$) have been explored experimentally at the ASDEX Upgrade tokamak (AUG), which provides a unique diagnostic suite, consisting of both high [2,3] and low field side valves and edge main ion CXRS systems. Using the D spectra, the molecular contamination is kept under tolerable levels in comparison to standard impurity measurements. Experimental and synthetic measurements quantitatively show that the Dα line is two orders of magnitude brighter than the Dγ line, which provides flexibility for optimized diagnostic setups.

The feasibility of the technique is explored through forward modeling, employing the FIDASIM4 code [4]. This study characterizes signal levels with respect to plasma density and temperature and gas puff flow rates, as well as the radial range in which the signal-to-noise ratio is sufficiently high. The modeling reveals that DCX and halo neutral populations are born in distinct radial locations and contribute to the measurement. Their radiation therefore contains information on the local ion temperature in the confined near separatrix (from the halo neutrals) and far-SOL regions (from the DCX neutrals), respectively. The emission of these neutrals gives a cold (~10 eV) and hotter (scenario dependent, 50-150 eV) contribution to the spectra. To validate the proposed diagnostic technique, the synthetic spectra are compared with first measurements at AUG. Care has been taken to include the Zeeman and instrument function corrections, which are crucial for the low temperature measurements. The approach proposed here not only addresses the persistent challenge of high-accuracy measurements of Ti at the separatrix, but can also provide information on the ion temperature in the SOL, to constrain the decay of the Ti profile into the far-SOL.

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SUPERCRITICAL CARBON DIOXIDE IMPINGING JET AS A COOLING AND HEAT RECOVERY MEDIUM IN PEBBLE BED BREEDING BLANKET FOR FUSION REACTORS

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The solid breeding blanket in fusion reactors plays a critical role in heat removal and energy recovery under extreme operational conditions. Currently, helium at 8 MPa and 300℃ is employed as the cooling and heat recovery medium. While helium, an inert and non-flammable gas, ensures safety in such environments, its low density necessitates significant pumping power to achieve the required mass flow rate and heat removal performance. This study numerically investigates the feasibility of using supercritical carbon dioxide (sCO_2) as a replacement for helium under identical pressure and temperature conditions in the impinging jet cooling of solid (pebble bed) breeding blankets. The cooling performance was evaluated using the Nusselt number as a primary indicator of heat transfer performance. Additionally, the pumping power required to achieve equivalent Reynolds numbers for both helium and sCO_2 was compared. To further assess the practical application of sCO_2, a performance metric considering equal pumping power was introduced to evaluate heat exchange efficiency. The results demonstrate that sCO_2 exhibits superior heat transfer capabilities compared to helium under similar conditions, with a significantly reduced pumping power requirement. This improvement is attributed to the higher density and specific heat of sCO_2, making it a promising alternative for enhancing the thermal performance and energy efficiency of solid breeding blankets.

Neutronics analysis of the blanket for a Plasma-Jet-Driven Magneto-Inertial Fusion Reactor

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This paper concentrates on the neutronics analysis for a hypothetical fusion reactor based on the repetitively pulsed concept of plasma-jet-driven magneto-inertial fusion (PJMIF).Taking PJMIF as the basis and referring to domestic and international fusion reactor breeding blanket technologies, a conceptual design of a 300 MW magneto-inertial confinement fusion reactor blanket was established, which adopted a Water-Cooled Lithium Lead (WCLL) structure. The WCLL relies on pressurized water as coolant and eutectic alloy Li17Pb83 enriched at 90% in 6Li as a tritium breeder, neutron multiplier and tritium carrier. In order to verify the rationality and feasibility of the WCLL blanket design scheme, the neutron transport process generated by fusion was simulated by using the three-dimensional Monte Carlo particle transport program MCNP, and a cloud map of the neutron flux distribution of this reactor was successfully drawn. Energy-dependent flux tallies are used to calculate neutron flux inside the blanket and outer wall, as well as the cylindrical ports where plasma guns are located. Tally multipliers of the flux in MCNP was used to estimated tritium breeding ratio (TBR) and nuclear heating. The TBR was calculated by taking the tritium produced in a given simulation and dividing it by the number of source neutrons emitted. The tritium enrichment ratios at different locations of the Li17Pb83 blanket were calculated and analyzed, and the sensitivity analysis and parameter optimization of the relevant factors affecting the tritium enrichment performance of the blanket, such as the first wall material, the number of plasma guns, the thickness of the enrichment zone, the blanket-to-spherical-center distance, and the 6Li enrichment, were carried out. This paper also explores the relationship between neutron flux and blanket thickness, and investigates neutronics properties such as neutron wall loading, nuclear thermal deposition, irradiation damage, etc., which provides a theoretical basis for evaluating parameters such as gun lifetime and dose rate of the shut-down reactor.

Overview and Design Development of CXRS-Pedestal Diagnostics System for ITER

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Indian Domestic agency (IN-DA) is developing a Charge eXchange Recombination Spectroscopy (CXRS) diagnostic system for ITER to diagnose ITER-pedestal region at 20 different radial locations with 20mm spatial resolution for measuring edge-ion temperature, impurity concentration and plasma rotation profiles at every 20 to 100 ms, whenever needed. The measurements of CXRS in pedestal region (CXRS-P) plays an important role in investigating physics of ITER-Pedestal region. For the conceptualization of CXRS-P diagnostics, the Simulation of Spectra (SOS) code [1] is extensively used. The synthetic signals predicted for different ITER scenarios and instrumental settings were used for optimizing the diagnostics setup and also line of sight geometry. The CXRS-P diagnostics system consists of Light Collection System (LCS), image Mis-Alignment Compensation system (MAC), Light Transmission System (LTS) and MultiBand MultiTrack (MBMT) spectrometer. The light from diagnostic neutral Beam & plasma interaction zone, is collected by LCS includes the set of lens and mirrors and transmitted via LTS. The LTS consists of ~220 fibers, distributed in a six fiber bundles assembly (FBA). The design is optimized in such a that five FBAs and five spectrometers can be used to cover 20 channels. The sixth bundle is used for calibration purpose. Each bundle consists of a group of 40 fibers arranged in a 4×10 matrix, where a group of 10 fibers represent a spatial location from plasma. The customized spectrometer can detect three emission wavelengths (two for plasma impurities (one dedicated for He-ash) and one for Beam emission), simultaneously. A spectrometer (~F#3) with a spectral resolution of ~0.4Å is proposed to couple light from 4×10 fiber bundle to a large 2D sensor (>2k × 2k). A sCMOS is considered as detector which can provide a time resolution of 20 ms. As a prototype activity, A 4 channels fiber bundle is already developed and tested for its transmission. A transmission of ~ 80% is achieved for bundle to bundle coupling including all terminations. This paper describes the present design of CXRS-P diagnostics system & subsequent R&D development and experimental results at ITER-India lab for meeting ITER requirements.

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Overview of Modeling for FLiBe Fusion Tritium Breeding Compatibility

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There is increased interest in the molten salt FLiBe ($2LiF\text{-}BeF_2$) as a tritium breeder for a fusion pilot plant, mainly due to its low electrical conductivity: it is less influenced by high magnetic fields and has a smaller impact on tokamak plasma control systems. A critical challenge for this blanket concept is the lack of compatible low-activation material; FLiBe melts at 459°C, too high for operation with reduced activation ferritic/martensitic (RAFM) steel (550°C maximum). Corrosion is also a concern; nickel alloys have superior corrosion resistance in salt but are not feasible in a fusion reactor due to their neutron activation characteristics, which lead to long-lived radioactive waste and decay heat production exceeding fission reactors. This work discusses a summary of results from a two-fold design strategy to address these shortcomings: 1) an alternate salt mixture (52% LiF/48% BeF_2) that melts at a lower temperature of 392°C, and 2) thin Ni, W, and Ni-W alloy coatings on RAFM as a means of corrosion mitigation. Project highlights the overview of 1) the thermal design of a FLiBe/RAFM blanket to identify an operating temperature window, using computational fluid dynamics and magnetohydrodynamics with optimization; 2) use neutronic and activation analyses to determine the impact of coating materials and salt composition on decay heat and waste production and the tritium breeding ratio, and 3) a series of capsule experiments in which coated RAFM specimens will be exposed to FLiBe to demonstrate their performance and validate thermodynamic-kinetic corrosion models.

This work summarizes the feasibility of a steady state thermal design for both first wall cooling and bulk breeding blanket with the 48 mol% BeF_2 composition of FLiBe with a RAFM steel channel structure. The study uses an analytical approach alongside simulations in STAR-CCM+ for thermal and hydrodynamic analysis to study the local and global effects as well as perform optimization studies to identify a design window for varying heat fluxes. The magnetohydrodynamic effects are also studied for their impact on the heat transfer and fluid results for up to 10 T magnetic field. Neutronic tools MCNP 6 and FISPACT II are used to evaluate the performance of the overall blanket concept design with coatings of Ni, W, and Ni-W for decay heat and waste disposal ratings. The coatings on the RAFM steel are considered in the feasibility study for the impact to heat transfer, decay heat and waste, and their ability to mitigate corrosion for the design and conditions.

Surrogate Modeling for Plasma Shape Control in MAST-U: Accelerating Tokamak Operations with AI

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Plasma shape control is a critical aspect of tokamak operations, directly influencing plasma stability, performance, and the success of advanced experimental campaigns. Traditional systems, such as EFIT, provide high-accuracy magnetic reconstructions that serve as preferred standard for plasma analysis. However, EFIT reconstructions are computationally intensive and performed post-shot, rendering them unsuitable for real-time applications [1]. The MAST-U plasma control system currently employs the Local Expansion MAST-U Reconstruction (LEMUR) algorithm for real-time monitoring and boundary reconstruction, leveraging a local expansion of poloidal flux constrained by the vacuum magnetic field equation [2, 3]. While LEMUR achieves sub-millisecond computational performance, its real-time estimates are considered less accurate than EFIT and limited to certain control variables [4]. In this work, we present a data-driven surrogate model designed to approximate EFIT reconstructions in real time, offering both high accuracy and computational efficiency.

Leveraging a dataset of magnetic signals, plasma currents, and EFIT-derived plasma shape parameters extracted from MAST-U shot data, we developed a feedforward neural network (FFNN) tailored for near real-time inference. The model was trained using a dataset encompassing multiple operational scenarios, including a variety of plasma configurations and dynamic behaviours. Preprocessing steps, including dimensionality reduction through Principal Component Analysis (PCA), ensured the retention of variance while reducing input features, facilitating model simplicity and computational efficiency.

The trained surrogate model accurately predicts key plasma shape parameters, such as x-point and strike-point location, within milliseconds, a marked improvement compared to conventional methods. Validation on unseen shot data highlights the model's generalisation capability, achieving a high correlation with EFIT-based reconstructions while operating orders of magnitude faster. This performance opens the door to applications such as real-time plasma shape feedback, advanced control strategies, and predictive scenario planning. By offering a fast and flexible alternative to LEMUR, the surrogate models have the potential to provide EFIT-like accuracy in real-time settings, addressing current limitations and enhancing the operational framework.

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In-Vessel Top Launch Electron Cyclotron Current Drive with a Graphite Refocusing Mirror at the DIII-D Tokamak

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A prototype Top Launch system has been developed at the DIII-D tokamak which utilizes an in-vessel focusing mirror to refocus an outside (low-field-side) launched EC beam into the plasma with a large polar angle and nearly-parallel path along the resonance line. The DIII-D EC system currently has eight steerable outside launchers and one of these will be used to test this prototype system. In 2020 and 2022, two top launch systems were installed on DIII-D using in-vessel waveguides through upper ports on the machine. These systems demonstrated the capability of top launch to drive higher off-axis ECCD and allowed DIII-D to achieve Advanced Tokamak relevant scenarios. However, one of the in-vessel waveguides failed due to an unknown runaway arcing condition and, as a result, both systems were removed in 2023. This new prototype is designed to replace the previous systems with a low-cost alternative that cannot fail in the same mode as previously. The prototype in-vessel mirror will be made of ATJ-grade graphite to match the material of the DIII-D first wall. It will be mounted in place of a tile at the top of the DIII-D vessel and use a section of an off-axis paraboloid to focus the beam. The mirror's surface is canted at a fixed angle to reflect the EC beam from the outside launcher at the desired angle. The mirror focuses the 110 GHz EC beam to a waist diameter of approximately 2.7cm at approximately 65cm from the mirror center. This was designed using a quasioptical model for a 110 GHz beam and will be measured as-installed via an infrared camera and a target setup. The mirror also has thermocouples embedded in it to sense the temperature of the tile and the approximate position of the beam. This prototype is predicted to demonstrate a current drive efficiency of about ζ_ECCD=0.2 when injected in an H-mode plasma. The mirror is predicted to reflect approximately 95% of the power incident on its surface, based on measurements performed with a vector network analyser on ATJ-grade graphite. If the resulting system is successful it will allow for the return of top launch electron cyclotron current drive at the DIII-D tokamak and can be iterated and duplicated, resulting in a top launch mirror for all 8 of the existing outside launchers at DIII-D.

Transient MHD Analysis of Liquid Metal in Tritium Breeding Blankets for Spherical Tokamak Reactor

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The Spherical Tokamak Advanced Reactor (STAR) is a compact low aspect ratio fusion reactor, defined with cost-effective features that collectively may lead to an economically viable fusion power plant design. One of the crucial components of this reactor is the tritium breeding blanket, which protects the reactor from neutron radiation, extracts heat from the fusion process and generates tritium. The blanket comprises of two sections: the in-board and out-board blankets. Both sections perform similar functions, However, the out-board blanket plays a more critical role due to its exposure to significantly higher neutron flux, which enhances its tritium breeding capacity. Its efficient performance is vital for achieving fuel self-sufficiency in fusion reactors by enabling tritium production. Among various blanket concepts explored for the STAR design, the Dual Coolant Lead Lithium (DCLL) blanket has demonstrated superior performance [1].

In this study, we have conducted a computational analysis of the DCLL outboard blanket to evaluate its thermal-magnetohydrodynamic (MHD) performance via customized ANSYS CFX 2024 R2 solver. The DCLL concept employs two coolants: eutectic PbLi and helium. Eutectic PbLi serves as the primary coolant, flowing through the main channels, with lead acting as the neutron multiplier and lithium functioning as the tritium breeder. Whereas Helium acts as the secondary coolant, primarily cooling the first walls and blanket structure, flowing along the outer side of the channels. The blanket's structural wall is made up of EUROFER 97, a Reduced Activation Ferritic/Martensitic (RAFM) steel, chosen for its ability to withstand extreme heat and radiation while minimizing neutron activation [2]. The study examines the MHD flow of liquid metal in the presence of a magnetic field, calculated using the Biot-Savart law. Our customized solver will be coupled with M3DC1, a specialized tool for solving 3D transient MHD equations in plasma on axisymmetric geometries [3]. Surface heating from charged particles and volumetric heating from neutronics, generated within the liquid metal coolant by interactions with energetic neutrons from the plasma, are also considered. Buoyancy effects in the primary coolant domain, caused by temperature gradients, further alter the MHD profile, affecting heat transfer from the liquid metal to the helium-cooled walls. Understanding the impact of key flow parameters on the heat transfer coefficient under these complex conditions is essential for optimizing the DCLL blanket design. Therefore, MHD flow and heat transfer analysis are critical aspects of designing virtual prototypes for fusion reactors.

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Overview of the in-vessel diagnostic systems for the SPARC tokamak

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SPARC is a compact, high-field, DT tokamak that is currently being built by Commonwealth Fusion Systems (CFS) in Devens, Massachusetts. The goal of SPARC is to demonstrate Q > 1 and derisk a path to the future powerplant, ARC. To support this mission, a suite of diagnostics has been designed with the intention of providing valuable data to support the safe operation of SPARC as well as determine a minimum viable set of diagnostics for ARC. The subset of the diagnostic systems within the primary vacuum boundary and embedded in divertor cassettes and port plugs represent the “in-vessel” diagnostics (INVD) of SPARC. These sensors include those for magnetic equilibria reconstruction and observing MHD activity, neutral pressure and gas analysis, resistive and optical bolometers for radiative power, as well as those used for boundary measurements like Langmuir probes, thermocouples, and fiber Bragg gratings. An overview of the complete SPARC diagnostic set has been published [1], as well as detailed papers on the individual in-vessel systems as they were in the early final design stage [2,3,4,5]. All systems have been designed to survive the challenging combination of disruption loading at elevated temperatures, while not violating the design constraints imposed by the ultra-high vacuum, tritium, and radiation environment. In addition, the INVD sensors located in the primary vacuum but outside of port plugs are not intended to be serviceable after the initial install. This requires either a robust design that will last the lifetime of SPARC or a sufficiently long life to gain the understanding that would allow operation without them. The in-vessel diagnostic systems are in the process of final design closeout, with many systems well into the procurement process. This work reviews the final designs, procurement progress, and the pre-assembly of SPARC parts ongoing at the CFS site in Devens, MA.

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ICRF Modeling in mirror and dipole geometries

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We are adapting the all-orders spectral algorithm (AORSA) [1] from tokamak geometry to axisymmetric magnetic mirror geometry. We will use the modified AORSA to study ion cyclotron range of frequency (ICRF) heating in the Wisconsin HTS Axisymmetric Mirror (WHAM) magnetic mirror device [2] and the OpenStar levitated dipole [3]. The ICRF power will be used in WHAM to accelerate high-energy neutral beam injection ions by ion cyclotron absorption at the second to fourth harmonics of deuterium. At these harmonics, full wave simulations most accurately capture wave propagation, including transmission, reflection, and absorption at the cyclotron layer. The antenna is modeled as a single-strap with m = 0 excitation, where m is the azimuthal mode number. We present some preliminary results for wave coupling and heating. The setup for OpenStar is similar. In both devices, there is no (or very little) toroidal field and several harmonics close to HTS coils.

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The hard X-ray monitor for SPARC: measuring MeV range photons in high neutron and

magnetic backgrounds

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The SPARC tokamak [1] will have a set of diagnostics monitoring the formation of dangerous relativistic, runaway electrons (REs) during plasma start-up. These REs can prevent plasma formation and damage plasma facing components of the machine. One of the diagnostics responsible for machine protection will be the hard X-ray (HXR) monitors [2]: two LaBr3 inorganic scintillators with a wide field of view embracing two different toroidal sections of SPARC. The detectors are responsible for measuring HXR with energies greater than 0.1 MeV emitted by start-up REs undergoing bremsstrahlung reactions with plasma ions and the tokamak first wall. Due to its position the detector will operate in an unprecedented harsh environment, with high neutron fluxes (~5 x 1010 n/cm2/s) and high residual magnetic fields (~100 Gs) from the poloidal field coils that could temporarily paralyze the detector or even burn its photomultiplier tube. In this work, we discuss the technical implementation of the HXR monitor with a simultaneous current and spectroscopic mode digital acquisition. A prototype has been characterized at MIT in terms of its energy resolution, efficiency, and dynamic range. We then present detailed neutronics simulations performed with OpenMC [4] and MCNP [5] for estimating the prompt neutron-induced background on the detector. FISPACT [6] simulations are also conducted to evaluate the delayed activation-induced background during SPARC operations. Based on these simulations we study different materials such as high density polyethylene (HDPE) and LiH as dedicated neutron attenuators to ensure the correct functioning of the detector throughout the first three SPARC campaigns. Finally, we conclude presenting our strategy to minimize and correct the effect of the external magnetic fields by a combination of mu-metal shielding and an LED-based monitor for the detector gain.

Studies of Hydrogenic Pellet Erosion and Impacts in a Gas Gun Type Pellet Injection System

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Pellet injectors based on continuous extruders that cut and accelerate pellets by high pressure gas in a gun barrel have been employed on magnetic fusion experiments for plasma fueling. This plasma fueling technique is planned to be used on the ITER burning plasma experiment and future fusion pilot plants. Recent measurements of H2 pellet erosion in the gun barrel of a gas gun injector were performed that showed a surprisingly strong erosion level and dependence on pellet speed [1]. This has motivated further experiments with H2, D2, and HD pellets accelerated in a gas gun to elucidate the erosion cause and dependence on injector parameters. Erosion of fuel material from cryogenic pellets needs to be minimized in a reactor system to maximize the fueling efficiency of the injector system and to minimize the accumulation of pellet material into the propellant gas that has to be recirculated in the injection system.

In addition to the erosion measurements that are made with microwave cavity diagnostics, a set of small impact guide tubes has been implemented to understand pellet survivability from multiple impacts. Future injection systems will have a number of jumps across diagnostics and valves that the pellets must survive. Each jump can lead to a small angle impact due to dispersion of the pellets when exiting a guide tube. Up to 6 impacts less than 2 degrees each are implemented in the injection line and pellet integrity is measured as a function of pellet species and speed. D2 pellets are found to have a higher probability of survival intact than H2 pellets. Implications of these results on fueling by pellet injection systems for burning plasmas and a fusion pilot plant are discussed.

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Keywords: Pellet injection, cryogenic extrusion, plasma fueling, hydrogen isotopic mixtures

Vertical stabilisation and control of the MAST-U tokamak via non-magnetic state estimation.

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A novel method of reactor-relevant vertical position control on the MAST-U tokamak is presented, using heat and particle flux balance to estimate plasma vertical position, in place of conventional magnetic diagnostics. Future reactor-scale tokamak designs such as STEP and ARC will exploit high-temperature superconducting magnets to achieve a compact device footprint, reducing size and cost. The intensified exhaust challenges in such devices necessitate advanced divertor configurations, including a double-null (DN) plasma shape, enabling power sharing between upper and lower divertors, mitigating heat flux on plasma-facing components and enhancing access to detachment.

Neutron irradiation from burning plasmas makes it untenable for magnetic sensors to be placed close to the plasma, severely limiting their viability as a state estimator for vertical stabilisation. Additionally, integral drift introduces a position measurement error which accumulates over long pulse lengths. This must be addressed as DN configurations require robust and precise vertical stabilisation as even minor deviations (~mm) in vertical position from a connected DN can result in significant divertor power imbalances, risking severe damage to plasma-facing components (D. Brunner et al 2018 Nucl. Fusion 58 076010). While DN operation poses a significant challenge in terms of control precision, it also presents an opportunity for magnetics-free sensing by using the power balance between the two divertors as the controlled quantity.

This contribution details the design, implementation and experimental validation of such a control scheme on MAST-U. Candidate MAST-U divertor kinetic diagnostics (Langmuir probes, Bolometry and D-alpha) are assessed for their suitability, with D-alpha selected for its reactor relevance and high temporal resolution. The upper and lower D-alpha signals are compared via an FPGA, forming an asymmetry signal which is passed to the vertical control system wherein a proportional derivative gain feedback controller actuates the radial field coils to achieve a desired asymmetry value.

Prior to experiment MATLAB/simulink feedback control simulations using D-alpha asymmetry data from prior experiments are conducted, providing rudimentary system identification prior to experiment. Step response comparison between non-magnetic and conventional vertical control schemes are conducted over a range of step amplitudes in MAST-U H-mode DN super-X plasmas. Up/down power balance performance is assessed for both control schemes, and alternative D-alpha inner strike point lines of site are investigated. These experiments lay the foundation for robust and precise magnetics-free vertical stabilisation and control in future reactor scale devices.

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Optimization of Square Duct Flow for Blanket First Wall Cooling: Tuning Generalized K-Omega Turbulence Model by Adjoint Optimization and Machine Learning

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Tuesday Posters 2, Lobdell (Building W20 Room 208), June 24, 2025, 2:00 PM - 3:30 PM

Helium cooling is widely considered for use in the plasma facing first wall (FW) of fusion reactors since it is chemically inert, non-radioactive, transparent to neutrons, and compatible with safety systems. In the CFD study of helium cooling components, Reynolds-averaged Navier-Stokes (RANS) modeling is popular due to the low cost and ease of use. However, RANS models are well-known for their lack of accuracy for prediction of complex flows. Previously, the heat transfer of a one-side-heated channel flow with ribs on the heated side was studied, and the errors introduced by RANS modeling were quantified. In addition, it was demonstrated that the Generalized k-ω (GEKO) model developed by ANSYS can be tuned to provide more accurate solutions.

In the work presented here, the adjoint method and machine learning via neural network (NN) are implemented to optimize the GEKO model. The heat transfer of a square duct flow is simulated and optimized using Direct Numerical Simulation (DNS) data as the ground truth. The bulk Reynolds number ranges from 4410 to 84000. The turbulence model is trained at the Reynolds number of 40000 and validated at Re=4410 and 84000. Comparisons with the DNS data are made before and after the turbulence model optimization.

xtrEMsense: sensors for the extremes of fusion

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Tuesday Posters 2, Lobdell (Building W20 Room 208), June 24, 2025, 2:00 PM - 3:30 PM

3-Sci Ltd, supported by United Kingdom Atomic Energy Authority (UKAEA) under the Fusion Industry Programme (FIP) Challenge, has conducted pioneering research in a novel electromagnetic-based technique to develop a suite of sensors, xtrEMsense©, for fusion environments. With a patent application filed, xtrEMsense© was initially aimed at monitoring temperatures of plasma facing components but recent adaptations have also resulted in a magnetic field sensing variant of the product. We report on latest results and capabilities of the xtrEMsense© sensors, and the benefits this novel technology is set to bring to the fusion industry.

Obtaining reliable temperature measurements from plasma facing components is critical to ensure safe operation and to aid efficient, effective operation and maintenance tasks. Understanding which components have endured excessive heat cycles or exceeded thresholds will help prioritise maintenance work, reducing power plant down time and provide reassurance that components remain fit for purpose. However, acquiring data in a fusion environment is challenging. Sensors must be capable of withstanding extreme high temperatures, be neutron resilient, and be immune to the electromagnetic environment.
xtrEMsense© is 3-Sci Ltd's answer. Based on electromagnetic techniques, xtrEMsense© provides sub°C resolution temperature measurements along a single elongated device from multiple sensing zones acting simultaneously and independently to provide a clear temperature profile of plasma-facing fusion components. The key benefits of xtrEMsense© are:

- Ultra-high temperature
- Inherent neutron resilience
- Distributed sensing
- Remote excitation and interrogation
- Single electrical breakthrough
- Robust design
- Designed with integration and compatibility with fusion structures in mind
- Fusion technology agnostic and highly adaptable

A combination of finite-element modelling, analytical calculations and extensive testing has been conducted to characterise the electromagnetic technique underpinning xtrEMsense© and optimise the design of the device itself. Key results to date include:

- xtrEMsense© can be made from any electrically conducting material and testing to date has included tungsten, stainless steel, copper and aluminium.
- The sensor is typically of the order of centimetres in cross-section and metres in length, highly configurable to meet user requirements.
- To date thirteen sensing zones, have been monitored simultaneously on a single device.
- Temperatures approaching 1000°C have been demonstrated so far.
- Resolution of sub°C has been achieved.
- Accompanying signal generation electronics and software user interface developed to support the first commercial products.

The xtrEMsense© products are designed to become a key enabler to the success of fusion as a commercial energy source through provision of data such as the temperature of plasma facing components and magnetic field strengths. The sensors' highly adaptable and configurable construction allows 3-Sci to develop further the xtrEMsense© products to match the requirements of the fusion industry and be suitable for all fusion reactor designs

The UKAEA PbLi Loop: Design Basis and Experimental Plan to Bridge a Gap in the Development of Liquid Metal Blankets

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Tuesday Posters 2, Lobdell (Building W20 Room 208), June 24, 2025, 2:00 PM - 3:30 PM

Breeding blankets are essential for achieving tritium self-sufficiency in commercial fusion power plants. Liquid metal blankets are a promising solution due to their dual role as tritium breeding materials and heat transfer properties. However, the use of electrically conductive, high-temperature fluids in fusion reactors presents unique challenges, particularly under strong magnetic fields. To address these challenges and advance the technology readiness level (TRL) of liquid metal blanket concepts, UKAEA is planning construction of a PbLi loop integrated with the CHIMERA test facility. This PbLi facility, which is under engineering design, will replicate fusion-relevant conditions by exposing PbLi flows to complex geometries, high temperatures and transient magnetic fields. By targeting dimensionless Hartmann, Reynolds and Grashof numbers that are critical to magnetohydrodynamics (MHD), the resulting experimental data can be directly applied to full-scale fusion designs. The results will support continuous blanket design and development, validate numerical models and inform risk mitigation strategies.

The PbLi facility prioritises scalability, modularity and the use of off-the-shelf components to reduce construction time and costs. Innovations in equipment, standardised design and assembly processes aim to benefit future experimental setups and fusion power plant services. The facility will support both steady-state and transient testing, providing critical insights into blanket performance during normal operation and fault scenarios. These efforts will aim to increase the TRL of liquid metal breeding blankets and refine reactor designs by optimising coolant flow paths, diagnostic placement and failure point identification.

Preliminary modelling of the Water-Cooled Lead Lithium (WCLL) breeding zone confirms the facility can achieve Hartmann, Reynolds and Grashof numbers on the same order of magnitude as those expected in EU DEMO, validating its relevance to near-term fusion reactors. These findings guide sensor placement, measurement strategies and the development of new diagnostic techniques for harsh PbLi environments.

The UKAEA PbLi Loop and CHIMERA test facility closes the gap between current liquid metal research and the operational requirements of future fusion reactors. By observing key MHD phenomena and using these to inform critical engineering correlations and other design tools under relevant fusion conditions, the facility provides a platform to accelerate the development of liquid metal breeding blankets. Outcomes from these targeted experimental campaigns will refine coolant flow paths, improve diagnostic capability and identify blanket design improvements, ultimately, bringing commercially viable fusion power closer to realisation.

Computational Error Field Validation Experiments on HIDRA

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Tuesday Posters 2, Lobdell (Building W20 Room 208), June 24, 2025, 2:00 PM - 3:30 PM

The Hybrid Illinois Device for Research and Applications (HIDRA) is a unique fusion device used for plasma material interaction (PMI) research at the University of Illinois at Urbana-Champaign (UIUC). HIDRA is the former WEGA stellarator that was operated at the Max Planck Institute for Plasma Physics in Greifswald, Germany. The machine is a five-period, l = 2, m = 5, hybrid tokamak-stellarator with a major radius R0 = 0.72 m and minor radius r = 0.19 m. The on-axis magnetic field of B0 = 0.0875 T can go as high as B0 = 0.5 T.

Computational work has been done to characterize the strength of the magnetic field and shape of the flux surfaces in this vacuum vessel based on different operational currents while running as a stellarator; the work has identified an error field possibly due to a misalignment of the central axes of the toroidal and helical coils. This error field creates non-periodic effects disrupting the five-period symmetry. Though the general shape of the plasma remains the same at symmetrical locations, the non-natural islands and the centermost point of the plasma, the magnetic axis, both move. The goal of this paper is to validate the magnitude and direction of this error field to ensure that our computational work agrees with the environment inside of HIDRA.

Toroidal coordinates were devised to define the magnetic field and the movement of particles for the computational work. The poloidal angle, the angle in a vertical cross-section of the torus, is given by θ. The toroidal angle, given by ϕ, measures the angular position on the torus and increases clockwise when viewed from above. This framework creates a right-hand coordinate system and ensures that the toroidal B-field generated by the magnets points in the + ϕ direction. Using this coordinate system, the experiment was designed to line up the computational ϕc=0 with the corresponding physical ϕ=0 on HIDRA. Without the error field, there would have been five equivalent options for this location and any one of them would have worked. However, with the error field's effects accounted for, there is only one correct location. In the code, the coil geometry has a positive helical coil crossing the midplane (θ=0) at ϕc=0, so the five symmetrical angles are based on where the physical positive helical coils cross the midplane.

Though there are many changes between symmetrical angles, the magnetic strength was used to determine the direction of the error field. A magnetic probe was designed to take magnetic field measurements between designated coil numbers; measurements were taken at multiple ports and at different distances into HIDRA. The code reported field strengths for the different symmetrical angles corresponding to these specific locations and the differences between non-symmetric angles were tabulated. These differences were then compared to the measured differences in magnetic strengths between port locations and used to find the correct placement of the computational ϕc=0.

Parametric and AI-Driven Optimization of WCLL Blanket Design for Enhanced Tritium Breeding

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Tuesday Posters 2, Lobdell (Building W20 Room 208), June 24, 2025, 2:00 PM - 3:30 PM

Water-Cooled Lithium Lead (WCLL) blankets employ liquid PbLi as the tritium breeder and pressurized water as the coolant. The excellent thermal properties and high tritium breeding potential of the liquid PbLi make WCLL blanket a key candidate for DEMO- scale fusion reactors. Designing a compact blanket shape with maximized tritium breeding ratio (TBR) is a critical objective for future commercial deployment. In particular, WCLL blankets often adopt the Single Module Segment (SMS) concept and are further divided into inboard and outboard segments based on proximity to the plasma. In this work, we propose a two-dimensional (2D) parametric approach that uses piecewise polynomial functions to model the inner and outer surfaces of the SMS-based WCLL blanket, thereby capturing the overall blanket geometry in a simplified yet flexible manner. This parametric representation benefits geometry generation efficiency and an overall systematic analysis of the blanket design. Based on OpenMC Monte Carlo neutron transport simulations, we calculated the TBR for various blanket configurations and determined the contribution of the PbLi volume correlates with unit tritium generation. By analyzing the PbLi consumption and integrating blanket thickness, we estimate the total blanket geometry.
To accelerate the design iteration, we develop an artificial neural network as a surrogate model, learning the nonlinear mapping from blanket geometry parameters to the simulated TBR. Combined with a genetic algorithm, this surrogate model efficiently explores the high-dimensional design space, searching for the optimal shape that maximize TBR while minimizing blanket size. Our method can significantly reduce computation time and enables automatic global blanket size optimization. These findings help guide the engineering feasibility of WCLL blanket designs, offering valuable insights into their overall geometry, tritium breeding performance, and liquid PbLi resource utilization.

Seeded Impurity Routing Sensitivity and Purge Study

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Tuesday Posters 2, Lobdell (Building W20 Room 208), June 24, 2025, 2:00 PM - 3:30 PM

To support heat management and transport, fusion powerplants require introduction of seeded impurities. In the STEP tokamak the main seeded impurities injected in the core and divertor will be xenon and argon, respectively. Careful control of injection rates is required to ensure effective detachment control of the plasma in the divertor.

While experiments on MAST and MAST-U coupled with plasma modelling have advanced understanding in how seeded impurities will impact plasma heat transport and detachment control, the understanding of how seeded impurities will impact the fuel cycle has been previously unexplored for a spherical tokamak. Once injected into the plasma, seeded impurities will become activated due to neutron capture resulting in formation of exceedingly small quantities of transmutation products. Further research is required to de-risk and develop future powerplant tokamak exhaust systems, and the STEP Fuel Cycle team are in the process of characterising and responding to the challenges posed by the handling of neutron-activated seeded impurities and their transmutation products.

The STEP Fuel Cycle team are using FISPACT-II modelling to understand the likely activation and transmutation products that the fuel cycle will be exposed to and will need to process. This will ensure at the earliest stages of design that protection of people, plant, and the environment is maintained aligning with ‘best available technique’ and ‘as low as reasonably practicable’ principles.

Findings are that the levels and activity of transmutation products formed is sensitive to seeded impurity routing through the tokamak. Additionally, cooling, and fractional purge strategies are feasible approaches for seeded impurity recycling and radiation management.

Real-Time Gas Temperature Monitoring via Frequency and Time Domain Reflectometry: Exploring novel tool for fast quench detection on HTS Magnets for Magnetic Confinement Fusion

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The results of numerical and experimental validations of fundamentals of the real-time quench monitoring technique based on frequency and time domain reflectometry (FTDR) are discussed. The methods for detection and analysis of microwave electromagnetic signals propagating in coolant gas channels have been recently proposed to monitor the temperature of the coolant and thus should be suitable for quench monitoring in superconducting magnets. The real time monitoring allows both observation of the quench evolution and its active prevention.
Through the studies which have been recently carried out, we show that variations of external conditions (temperature and/or pressure) induce changes in the gas permittivity, which can be accurately measured. One notes that these conditions can vary either due to heat release or cooling pipe degradation/damage and important for SC magnet “health” monitoring. Observing a range of a frequencies and analysing the spectral lines shifts is useful to investigate the average gas properties fluctuations within the whole system. Measuring variations of the amplitudes and the phases at a single frequency, allows evaluation of the gas local temperature fluctuations, i.e. either the hotspots appearance in the pipe or an anomalous behaviour of the gas flow.
The methodology presented is based on the established correlations between thermodynamic gas variables such as temperature and pressure with its electromagnetic properties i.e. refractive index. The technique has been subjected to both numerical and experimental validations. The numerical studies conducted demonstrated that a localised hotspot can be detected with minimal time delay. The experimental data observed agree well with the theoretical understandings and will be also presented. The findings demonstrate the validity of the proposed technique and its ability to provide a robust and non-invasive capabilities for the quench prevention and magnet health monitoring. While the technique holds significant potential for high-temperature SC magnets it also offers a valuable tool for researchers and engineers in this area encompassing many devices using such magnets, including fusion reactors, MRI systems and others.

Hollow Cathodes Discharges as an alternative for NBI Sources: preliminary results on the ATHENIS facility

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Hollow Cathodes (HCs) are a wide-used technology in Space Propulsion, representing nowadays the standard for Electric Thrusters both as neutralizers and plasma heating source. HC discharges have been proven to be highly efficient when compared to filament or RF discharges: as a matter of fact, their DC nature removes all issues related to inductive coupling - typical of RF discharges - while the demonstrated lifetimes of tens of thousands of hours without any maintenance make these devices a valuable alternative to typical arc discharges. Consorzio RFX in Padua and University of Pisa have been testing this technology on the ATHENIS facility, an innovative concept of a new NBI source based on Hall Thruster designs, both in Argon and with a mix of Argon and Hydrogen as feed gas, with plasma parameters characterized via a movable Langmuir Probe. The preliminary results, described in this paper, show good plasma densities (in the order of 1.0E17 m-3) even at low injected powers (less than 250 kW/m3), both in Argon and with the Argon-Hydrogen mix. Furthermore, in the mix fuel gas case, the Langmuir Probe characteristics indicate the presence of a second electron population, well described by a temperature similar to the accelerating potential of the HC-extracted electrons. The electron temperature remains in the range of 2-4 eV in all tested configurations, indicating that the radial magnetic field of the Hall thruster acted efficiently also as magnetic filter, representing a promising features towards negative ion production.

PARAMETRIC DYNAMIC MODE DECOMPOSITION FOR NUCLEAR FUSION: APPLICATION TO MAGNETOHYDRODYNAMICS

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Magnetohydrodynamics (MHD) investigates the dynamics of electrically conducting fluids interacting with magnetic fields. This theory is widely utilized for magnetic confinement fusion (MCF) problems since both thermonuclear plasmas and operating fluids flowing in the blanket can be described as conducting fluids subjected to magnetic fields. The intrinsically multi-physical nature of MHD problems leads to complex models of equations whose numerical resolution often results in high computational costs, making them unsuited for multi-query applications such as parameter optimization and sensitivity analysis, as well as for online monitoring and control.

Reduced Order Modelling (ROM) techniques constitute a promising solution to the trade-off between computational expenses and the desired model accuracy, as proven in various engineering fields. Despite this, their application for MHD problems, and in general within the nuclear fusion field, is still in its preliminary stages. At their core, all ROM techniques aim to retrieve a low-dimensional yet faithful representation of the starting model. Among all ROM methods, Dynamic Mode Decomposition (DMD) represents one of the most commonly used ROM methodologies: DMD is a purely data-driven method designed to learn the best linear representation of the provided time-series data.

In light of the above, this study continues the investigation started by the authors on ROM techniques for MHD problems by considering a novel parametric version of DMD, aiming to retrieve a general linear representation of the starting parametric time-series data. Compared to standard single-parameter DMD and the state-of-the-art version of parametric DMD, this novel approach considers the DMD operators as snapshots to obtain a map between parameter values and modal coefficients to retrieve the transient dynamics across several parametric instances, enhancing computational efficiency and providing generalization compared to the standard algorithm. The final goal is to obtain a general model capable of performing both interpolation and extrapolation of the given parametric domain, that is, to accurately compute the transient dynamics given a previously unknown realization of the given parameter.

As a test case, this work considers a lead-lithium magnetohydrodynamic flow subjected to a magnetic field of varying intensity. The results show that the parametric DMD can be a promising solution for the computational burden of MHD simulations in light of the drastic reduction in computational times whilst retaining acceptable accuracy levels. They highlight the potential of parametric DMD for general applications in MHD scenarios involved in fusion reactors.

Machine Learning-Enhanced Multimodal Super-Resolution Diagnostics with Application to ELM and RMP Mechanism Analysis

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We introduce a groundbreaking multimodal super-resolution diagnostic framework (Diag2Diag) based on machine learning, which enhances the temporal resolution of Thomson Scattering (TS) diagnostics at DIII-D from approximately 200 Hz to 1 MHz. This advancement facilitates unprecedented visualization of fast transient phenomena, such as Edge Localized Modes (ELMs), and provides direct experimental evidence of resonant magnetic perturbation (RMP)-induced temperature and density profile flattening.
Diag2Diag leverages hidden correlations among complementary diagnostics, including Electron Cyclotron Emission (ECE), interferometry, and magnetics, to generate synthetic super-resolution Thomson Scattering (SRTS) data. This innovation allows the direct observation of individual ELM events—including their onset, crash, and recovery—without the need for traditional cycle aggregation. For the first time, these events can be studied as discrete phenomena, yielding deeper insights into their dynamics and implications for mitigating edge damage in tokamaks.
Moreover, this framework delivers experimental validation of RMP mechanisms, such as the formation of magnetic islands at the plasma pedestal boundary, confirming long-standing theoretical predictions. The simultaneous flattening of temperature and density profiles observed through SRTS provides a critical breakthrough in understanding plasma behavior under the influence of RMPs. These findings hold significant promise for enhancing plasma confinement and mitigating transient plasma instabilities.
Beyond its diagnostic precision, Diag2Diag addresses longstanding challenges related to diagnostic reliability. By reconstructing missing or low-resolution TS data using correlations with other diagnostics, the method achieves an R2-score of 0.92. This capability ensures continuous measurement reliability, even in the event of diagnostic hardware malfunctions, aligning with the operational needs of future fusion reactors like ITER. Furthermore, the methodology enhances diagnostic capabilities without requiring costly hardware upgrades, making it a practical and efficient solution for advancing fusion diagnostics.
The impact of this work extends beyond tokamaks and fusion energy applications. The ability to enhance diagnostic resolution and reveal hidden physical phenomena without additional hardware has potential applications in fields such as astrophysics, medical imaging, and laser fusion. Furthermore, the insights gained from this research may lead to the discovery of new plasma states and a more nuanced understanding of the coupled dynamics of density and temperature gradients during transient events.
By addressing longstanding limitations in diagnostic techniques and enabling experimental validation of theoretical models, Diag2Diag represents a transformative step for experimental fusion science. This research not only advances our understanding of plasma behavior but also provides practical tools to support the development of sustainable fusion reactors. The approach bridges the gap between experimental constraints and the need for physics discovery, creating opportunities for significant breakthroughs in both fusion energy and other scientific domains.
This material is based upon work supported by the U.S. Department of Energy, Office of Science, Office of Fusion Energy Sciences, using the National Fusion Facility, a DOE Office of Science user facility, under Award DE-FC02-04ER54698. In addition, this research was supported by the U.S. Department of Energy, under Awards DE-SC0024527, DE-SC0015480, DE-SC0022270 and DE-SC0022272, as well as the National Research Foundation of Korea (NRF) Award RS-2024-00346024 funded by the Korea government (MSIT).

Precision Control and Diagnostics for Operating a Cryogenic Pellet Ablation Test Stand

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Cryogenic pellets, typically composed of hydrogen isotopes or noble gases, are essential for core fueling, edge-localized mode (ELM) pacing, and disruption mitigation in fusion devices. These pellets ablate into the bulk plasma, stabilizing instabilities and sustaining fusion reactions. A cryogenic pellet injection and ablation test stand has been developed at Columbia University in collaboration with Oak Ridge National Laboratory (ORNL), providing a controlled environment for studying ablation processes, refining injection models, and developing the technology of material injection. A first experiment will explore ablation physics, which is challenging to study in tokamaks due to diagnostic limitations and complex plasma dynamics. The test stand linearly propels cryogenic pellets into an electron beam, which acts as the ablation source. Diagnostics include a high-speed camera and microwave cavities for measuring pellet speed and mass. This contribution focuses on the implementation of a LabVIEW-based automation system central to the experiment's operation that supports both manual and automatic modes. This system manages “pellet recipes” by precisely regulating gas composition, flow rates, and temperature stabilization via PID control, ensuring consistent and reproducible pellet formation. The test stand has successfully produced and injected H_2 pellets, validating its operational readiness.

Multi-Photon Absorption Laser-Induced Fluorescence Techniques as a Particle Diagnostic for Nuclear Fusion Reactors

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Measurements of the absolute hydrogen neutral density in the edge of a fusion plasma are needed to provide the critical information necessary for real-time plasma fueling control, calculation of charge-exchange power losses, control of plasma-wall interactions, determination of the braking of plasma flow, determination of the fuel mixture in deuterium-tritium plasmas, and understanding the dynamics of the divertor region in the plasma edge. Neutrals play a critically important role in magnetically-confined fusion plasmas by providing a source of new plasma and a channel for heat transport across the confining magnetic field. Energetic neutrals rapidly escape the plasma and can sputter impurities into the plasma after impacting the walls of the device, a deleterious effect, or can even be used for charge exchange measurements of the bulk ion temperature, a useful side-effect. Measurements with laser-induced fluorescence (LIF) techniques provide spatially-localized plasma characteristics by changing the quantum state of bound electrons, without perturbing the interrogated atom. Bound-electron transitions in hydrogen require photon energies which are unfeasible for single-photon LIF given modern laser technology. Two-photon Absorption LIF (TALIF) requires photons at half the transition energy, and Three-photon absorption LIF (3pLIF) requires photons at one-third the transition energy. While multi-photon absorption LIF requires much higher laser powers than single-photon LIF to be feasible, the wavelength requirements are much more relaxed since H transitions at 102.5 nm are easily accessible at wavelengths of 205 nm (TALIF) and 307.5 nm (3pLIF). Pulsed laser systems provide the necessary power and wavelength requirements to measure the absolute hydrogen density at a cadence of around 1 kHz. Here we present two and three photon absorption LIF measurements of the absolute density of neutral hydrogen. Relative cross-sections are determined between hydrogen and krypton for the two-photon measurements, then the two- and three- photon hydrogen measurements are made for the same fill pressures to measure the relative cross-section of three-photon absorption of hydrogen to two-photon absorption of hydrogen and krypton.

Study on the dynamics of cold pulse induced by compact torus injection in EAST Tokamak

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This paper investigates the non-local phenomena induced by Compact Torus (CT) injection during the propagation of cold pulses in EAST tokamak plasmas, focusing on the spatiotemporal evolution of electron temperature, electron density, and their fluctuations. Non-locality implies a core response to edge perturbations on a timescale much shorter than the global diffusion model constraint time. CT is a high-density, high-velocity, and self-organized plasma formed and axially accelerated in a coaxial electrode, and the compact torus injection device is used to fuel the tokamak[1][2]. The EAST compact torus injection system has a wide operating range, with a velocity range of 100-300 km/s and an average electron density range of 0.2-1.2×10^{22} m^{-3}. Through CT injection experiments on EAST tokamak, it was observed that while the density increased, the core temperature of EAST also rose rapidly with edge perturbations[3][4][5]. By observing the spatiotemporal structure of electron temperature fluctuation signals in the plasma and using correlation analysis, the spatiotemporal characteristics of temperature fluctuations were revealed. In particular, the dynamic characteristics of remote fluctuations were studied, and it was found that during the non-local phase induced by CT injection, the radial length of inward-propagating temperature fluctuations was significantly enhanced. Utilizing the wide operating range of CT injection, research on non-local phenomena in tokamaks is of direct significance for understanding the dynamic interaction between turbulence and large-scale mode structures in fusion plasmas.

Overview of Port Plug Based UV-MIR Optical Diagnostic Systems for SPARC

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SPARC is a compact, high-field, D-T tokamak that is presently under construction and will be used to de-risk the high-field path to commercial fusion energy. The SPARC UV-MIR optical diagnostic systems are in the process of completing the design of the diagnostic port interfacing components within the vacuum boundary. These systems will be deployed on SPARC for real-time control, to verify scientific milestones, and provide crucial data for the design of the ARC powerplant. The diagnostics covering the optical UV-MIR wavelength range on SPARC presented here are Interferometry (INTF)[1], Core Thomson Scattering (CTST), UV, visible and IR Imaging (IMAG), and the UV and visible Spectroscopy (SPEC) systems. The work presented is focused the unique design for each of these diagnostics systems driven by the SPARC in-vessel environment, where diagnostic assemblies are exposed to high radiation flux, strong magnetic fields, EM forces, plasma surface interactions, and high temperature gradients. These systems also occupy a limited space that must be shared between diagnostic systems and the required in port radiation shielding within each of the SPARC port plug assemblies, further impacting the design of these systems. In specific, this work is highlighting the completed design of the INTF, TSCT, and IMAG systems inside the SPARC vacuum vessel as well as sharing details from the first fully assembled assemblies for these systems and the status of integration into a SPARC port plug. The full scope of these port plug assembly's is comprised of a combination of large metallic optics made of N-50 Stainless Steel, complicated optical enclosures, protective shielding, a laser beam dump array, and custom Sapphire / Silicon-Carbide optical assemblies for high power CTST laser mirrors. In this contribution the most impactful updates of these four diagnostic systems will be presented to highlight the path for these systems to ultimately provide the quality measurements required for plasma operations, validate SPARC performance towards Q > 1, and help inform the design of ARC[2].

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The Effect of Ion Irradiation and High Temperatures on the Mechanical Properties and Phase Stability of the 'Octalithium Ceramics' (Li8MO6, M = Zr, Sn, Pb and Ce) for Tritium Breeding

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In order to most efficiently produce tritium from a high energy neutronic reaction, lithium dense tritium breeding materials (TBMs) are required. TBMs must operate under high temperatures and neutron radiation, whilst producing extractable tritium and being compatible with the surrounding materials. Ceramic TBMs offer improved material compatibility and do not suffer from magnetohydrodynamic (MHD) effects, however, traditionally they have lower tritium breeding ratios (TBRs) in addition to concerns over radiation damage.

With the current industrial interest in spherical tokamak arrangements with less space for TBMs, materials with higher TBRs are required. Neutronics simulations suggest that the octalithium compounds, with their high lithium densities, offer significantly higher TBRs[1] than Li_4SiO_4 and Li_2TiO_3 which are designated for use in ITER – however most of these compounds lack basic physical data (melting points, phase stability, mechanical properties) and none have been subject to micro mechanical and ion irradiation testing.

This work presents the mechanical properties (Youngs modulus, hardness and fracture toughness) of dense octalithium ceramics (Li_8MO_6, M = Zr, Pb, Sn and Ce) from nanoindentation, how these experimental values correspond with those predicted using density functional theory modelling (DFT), and the impact of high energy (12 MeV, $1e^{17}cm^{-2}$) He ion irradiation on these properties. Further we examine how the octalithiums will perform in the hostile environment of a future reactor, by exploring the phase stability at high temperatures (500°C, 700°C and 900°C) using X-ray diffraction and mass loss.

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Numerical assessment on the impurity transport in the W7-X particle exhaust

using the Direct Simulation Monte Carlo method

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In a fusion device, divertor radiation via seeded impurities is crucial for spreading the incident power over a larger divertor surface area and achieving partially detached conditions, particularly in high-power-flux devices such as ITER or DEMO. However, impurity seeding introduces challenges, including fuel dilution in the core and potential degradation of energy confinement due to central radiative losses. Thus, achieving a high impurity enrichment specifically in the divertor region is essential. The neutral impurity concentrations in the divertor are influenced by variations in neutral conductances and pumping speeds. For typical W7-X sub-divertor conditions, H_2 molecules operate in a transition flow regime between the free molecular and viscous flow regimes, leading to an increase in the effective H_2 pumping speed with rising pressure. Neutral impurities (e.g., He, Ar, and Ne) are entrained in the primary H_2 bulk flow, which enhances their effective conductance [1].

Within this context, the present numerical study models the neutral transport of fuel gas (H_2) along with seeding impurities (Ar, Ne) and helium in the W7-X sub-divertor. This is accomplished using the Direct Simulation Monte Carlo (DSMC) [2] solver within the DIVGAS workflow [3,4]. The simulations aim to estimate key macroscopic parameters, such as partial pressure, temperature, and molar fraction for each gas species. Additionally, the study evaluates the entrainment factor - defined as the ratio of impurity velocity to molecular hydrogen velocity - which reflects the accelerated removal rate of impurities from the main fuel gas flow. This factor plays a vital role in reproducing the global impurity decay time [1].

The results provide valuable insights into neutral impurity transport in the W7-X sub-divertor and its implications for exhaust performance. These findings offer practical guidelines for optimizing the current design and could support direct comparisons with experimental data. This study, which focuses on large-scale 3D simulations, is the first of its kind. It serves as a fundamental step toward designing and optimizing gas exhaust systems for W7-X and other reactor-scale Stellarator devices. Furthermore, it has implications for related diagnostics, such as the Diagnostic Residual Gas Analyser (DRGA) currently installed in W7-X.

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Long-Pulse (1000 s) Adaptive X-Ray Technology for WEST: Physics Breakthroughs and Engineering Challenges

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Novel multi-energy soft and hard x-ray (ME-SXR & ME-HXR) diagnostics have been designed build and installed by an international collaboration in WEST (W Environment in Steady-state Tokamak), at the CEA site in Cadarache, France. These pin-hole cameras employ a pixelated 2D x-ray detector in which the lower energy threshold for photon detection can be adjusted independently on each pixel of the detector. The use these versatile Si and CdTe multi-energy systems provide unprecedented improvement in throughput and signal-to-noise-ratio enabling adequate spatial and time-resolution (dr/a~2%, dt~1 ms) as well as energy discrimination at complementary ranges of Ephoton~1-20×Te,0 and Ephoton~20-100×Te,0 for the SXR and HXR options, respectively. These diagnostics have provided spatialy and energy resolved measurements of soft and hard x-ray emission from the full plasma cross-section resolving impurity density measurements from the tungsten line-emission contribution, electron temperature measurements from the energy-resolved continuum emission, the emission from the fast electron tail density produced by the plasma startup, radiofrequency current drive and runaway electrons, as well as and the tungsten x-ray emission due beam-target emission at the tungsten walls. In this contribution we discuss the design and architecture of the ME-SXR and -HXR diagnostics, their integration to the WEST tokamak and the engineering challenges that the short (up to 60 s) and long pulse scenarios (up to 1000s) poses to vacuum interface, radiated power and electronics. Vacuum and thermal stress analysis and heat transfer calculations will be presented and discussed.

Preliminary hodoscope characterization for the magnetic proton recoil based neutron spectrometer of the SPARC tokamak

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The SPARC tokamak aims to demonstrate a net fusion power gain in tokamaks for the first time (Q-fus>1)[1]. A neutron spectrometer (NSPC) based on the magnetic proton recoil technique [2] is being built to measure neutrons with energy between 1 and 20 MeV emitted by fusion reactions. It will infer the total fusion power emitted by the machine, corroborating the demonstration of Q-fus>1 through the primary P-fus measurement from other neutron diagnostics [3]. This diagnostic will also provide useful information on ion temperature, D/T ratio, and non/thermal fusion neutrons.
The SPARC NSPC is based on the generation of recoil protons via elastic scattering of collimated neutrons on a target and their dispersion via a set of magnets [4][5]. The recoil protons are momentum analysed by the magnets and dispersed on a hodoscope consisting of an array of ~40 detectors made of EJ276 or similar plastic scintillators coupled to photomultiplier tubes [6]. The hodoscope is made of three groups of channels whose dimensions are optimized based on the ion optics and the energy of protons.
In this work, the performance of two prototypical EJ276 scintillation rods of size (1x1x11) cm^3 and (0.8x0.5x10) cm^3, respectively, are assessed by coupling two curved light guides using a silicone rubber optical interface to one or two photomultiplier tubes. The performances of these detector prototypes were measured with protons, alphas, neutrons, and gamma ray sources. Techniques to acquire the hodoscope data and study the spectrum collected by the hodoscope are also presented.

This work is supported by Commonwealth Fusion Systems and ENI SpA.

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Current status and future plans for research and development toward helical fusion pilot plant

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Tuesday Parallel 3a - Next Steps II, Kresge Main Theater (Building W16, upstairs), June 24, 2025, 4:00 PM - 5:30 PM

Helical Fusion Co., Ltd. (HF) is a fusion start-up company spun out from the National Institute for Fusion Science in Japan, aiming to realize a helical fusion power plant using continuously wound helical coils. HF is promoting the development of a steady-state fusion power plant based on physics and engineering basis that can be scalable to fusion power plants, which have been established through experiments with the Large Helical Device and fusion reactor design / fusion engineering research at the National Institute for Fusion Science over many years.
In order to realize more attractive power plants as soon as possible, HF is working on developing proprietary technologies. The superconducting magnet uses a proprietary conductor made of high-temperature superconducting tapes to achieve high current density, easy winding, and advanced quench protection. Tin-based liquid metal with low vapor pressure is used as the tritium breeder and coolant in the blanket, which enables highly efficient power generation by extracting high-temperature liquid metal, and protection of the surfaces of in-vessel components from high heat and particle loads. The blanket casing is made of non-magnetic reduced-activation material to avoid the influence of magnetization on the confinement magnetic field and maintenance work. The large aperture between magnets allows all blanket modules to be replaced only by crane from the upper ports, resulting in high plant availability. HF develops not only these two main components, but also all plant equipment, such as plasma heating system, cryogenic system, fueling and vacuum pumping system, diagnostic and control system, power generation system, and assembly and maintenance system.
Currently, individual demonstrations of these technologies are being conducted in collaboration with universities, research institutes, and partner companies. In the 2020s, a small experimental device equipped with all the components of a fusion power plant (Final Experimental Device, FED) will be constructed to demonstrate engineering integration and ultra-long-hour continuous operation. A fusion pilot plant (FPP) will then be constructed in the early 2030s and operated with a net electrical output of several tens of MW. The presentation will provide details on the latest status of these developments and future plans.

Overview of Magnet Zero

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Tuesday Parallel 3a - Next Steps II, Kresge Main Theater (Building W16, upstairs), June 24, 2025, 4:00 PM - 5:30 PM

Magnet Zero (M0) is a first-of-its-kind high-temperature superconducting (HTS) stellarator magnet prototype, designed and built by Type One Energy Group Inc. in collaboration with MIT's Plasma Science and Fusion Center. M0 serves as a test bed for developing and testing key technologies and processes needed for high-field non-planar magnets of relevant shape, size, and field for commercial stellarator power plants. M0 contains multiple coupled HTS magnet architectures, including auxiliary coils that increase the magnetic field at the non-planar regions with the tightest bends, testing operational robustness in fields of magnitude and direction representative of what would be experienced in a high-field stellarator. M0 is being tested at the MIT PSFC Superconducting Magnet Test Facility, which enables operation at high currents (10s kA) and low temperatures (~20 K). This talk describes the design and fabrication of M0, and preliminary test results.

Steady state, intrinsically stable fusion power plants: the QI-HTS stellarator approach

Sciortino F[1], the Proxima Fusion team[1]
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Tuesday Parallel 3a - Next Steps II, Kresge Main Theater (Building W16, upstairs), June 24, 2025, 4:00 PM - 5:30 PM

Stellarators have historically suffered from poor cross-field transport, low alpha particle confinement, and manufacturing challenges. However, experiments at the W7-X stellarator at the Max Planck Institute for Plasma Physics have conclusively demonstrated that optimized stellarators can reach performance analogous to tokamaks, while benefiting from disruption-free steady-state operation. Moreover, recent work [Landreman & Paul, PRL 2022] has shown that stellarators can be optimized to confine alphas even more effectively than tokamaks. Doing so in a precise quasi-isodynamic (QI) design [Goodman et al., PRX 2023] enables the avoidance of toroidal plasma currents, and therefore current-driven plasma instabilities. Compared to tokamaks, the absence of disruptions implies significant design simplifications, particularly in a commercially-relevant context of energy production. The prospect of steady-state, intrinsically stable fusion power plants is now realistic.

Additionally, advances in high-temperature superconducting (HTS) magnet technology have recently led to significant progress in tokamaks, enabling a major size reduction through operation at higher field. However, the application of the same HTS technology for non-planar magnets at the scale needed for stellarator reactors is yet to be demonstrated.

In this talk, we will introduce the concept of a QI stellarator and compare it to the more conventional tokamak. We will present the “Stellaris” concept [Lion et al., FED 2025], a stellarator designed for power plant applications, which simultaneously meets all key physics and engineering constraints for energy production. A simulation-driven engineering approach will be shown to lead to simplified stellarator designs that leverage manufacturing capabilities already de-risked by W7-X and other fusion experiments.

Requirements for Feasibility and Solutions for Low Hydrogen Recycling Liquid Metal Flow Systems in Commercially Attractive Fusion Reactors

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Tuesday Parallel 3b - Divertors and Plasma Facing Components II, Sala de Puerto Rico (Building W20 Room 202), June 24, 2025, 4:00 PM - 5:30 PM

Future commercial fusion reactors are expected to impose large heat deposition on plasma-facing components. Currently, solid plasma-facing components in fusion environments suffer from melting and erosion due to intense heat loads from reactor plasmas. Free-surface flows of liquid lithium are promising for mitigating these issues as they do not permanently reshape and they could offer almost immediate material replenishment after a plasma disruption. Moreover, lithium offers low hydrogen recycling, which allows higher plasma temperatures and it has been shown to improve confinement in fusion devices. However, there is not a consensus on the optimal configuration for a free-surface liquid lithium system in a fusion reactor. Basic liquid-metal-flow systems consist of linear chutes where bulk flows of liquid metal (LM) absorb heat and transfer it through advection. This configuration is attractive for its simplicity of construction but driving the flow at the required speeds (~10 m/s) to avoid overheating demands the following: (1) large liquid lithium inventories (safety concerns), (2) excessive power from external pumping to overcome magnetohydrodynamic (MHD) drag and (3) it puts free-surface flows at risk of piling and splashing. Most of the research on this type of systems has focused on plasma-material interaction with the free-surface flow and consequent instabilities instead of guaranteeing a stable bulk LM flow with the interaction of reactor-magnetic fields. It was up until recently that experiments at the Liquid Metal eXperiment of the Princeton Plasma Physics Laboratory (USA) and the NIFS Oroshhi-2 Superconducting Facility (Japan), and computer simulations with FreeMHD (recently developed code for free-surface LM simulations at the reactor scale) have addressed possible solutions to the aforementioned disadvantages. Currently, none of the proposed fast-flow-LM systems pass basic feasibility tests. Thus, a solution with engineering simplicity for a fast-free-surface-liquid-lithium flow under divertor conditions must have the following qualities to be commercially attractive: (1) small LM-flow speed to make LM-flow-volume minute and to reduce flow-pumping-power requirements, (2) operation with lithium to achieve low-recycling, (3) small size of LM inventories, (3) in-situ tritium distillation, (4) easiness of construction/installation (single material substrate/pipes for LM flow) (5) and no moving components for the LM system inside the reactor environment. Divertorlets is our proposal that satisfies all these qualities. Divertorlets operate with slow flows of liquid lithium and it has a moving free-surface exposed to the plasma for low-hydrogen recycling conditions. This configuration reduces the risk of impurity injection into the plasma for its stable free surface. It also allows the implementation of tungsten meshes at the free surface to improve stability. Current projections for operations indicate that divertorlets less than 10% of the power output expected for a fusion reactor while other alternatives have reported to require 30% and above, making it a very compelling configuration for reactor operation.

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Design and Analysis of Internal Cooling Geometries for High Heat Flux Divertor Components in Transient and Steady State Thermal Conditions for Use in The Infinity Two Fusion Pilot Plant Stellarator

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Tuesday Parallel 3b - Divertors and Plasma Facing Components II, Sala de Puerto Rico (Building W20 Room 202), June 24, 2025, 4:00 PM - 5:30 PM

There are many variables to consider when designing active cooling solutions for high heat flux components. The extreme environmental conditions and space constraints quickly approach the boundary of acceptable material limitations. In addition, specific systems requirements related to component interfacing must also be met. Due to these factors, a complex internal active cooling solution is needed that can be applied to both a risk reduction platform and subsequent fusion pilot plant. These complex geometries will require the utilization of available additive manufacturing techniques. This project focuses on the design and analysis of complex internal geometries for active cooling of high heat flux components, specifically a stellarator divertor. The design solutions are evaluated from a first principles basis to minimize mass flow rate of selected coolants while using the maximum material temperature as the boundary condition. This analysis is then used as the basis to build a parametric model of the internal cooling geometry. Next, the model is verified via a conjugate heat transfer simulation and followed by a parametric study to optimize the coolant flow based on the mass flow rate and allowable pressure drop. Finally, these solutions are then analyzed for both steady state and transient thermal loading scenarios to create a baseline cooling solution for the Infinity One Risk Reduction Platform stellarator divertor. This design will then be scaled accordingly to the Infinity Two Fusion Pilot Plant stellarator.

Article Testing for Scrape Off Layer Diagnostic in WEST

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Tuesday Parallel 3b - Divertors and Plasma Facing Components II, Sala de Puerto Rico (Building W20 Room 202), June 24, 2025, 4:00 PM - 5:30 PM

Steady-state operation of next-step fusion devices will require actively cooled Plasma Facing Components (PFCs). The extreme environmental conditions (high heat-fluxes, plasma interaction, erosion, transient loads, etc.) will also lead to complex geometries that require advance manufacturing techniques. Oak Ridge National Laboratory (ORNL) is collaborating with the CEA Institute for Magnetic Fusion Research (IRFM) in the development of a scrape off layer diagnostic for use in the W Environment in Steady-state Tokamak (WEST) in Cadarache, France. This diagnostic will include imbedded temperature sensors, as well as sample areas for ex-situ analysis of plasma-material interactions. The probe will be water-cooled and must transit into the scrape off layer for periods of several minutes, experiencing steady-state high heat fluxes (~3 MW/m^2) as well as extreme thermal and mechanical transient loads in the case of a plasma disruption. The need to translate ~0.5m into the vacuum chamber adds complexity to the design.

To survive the complex and conflicting requirements, the probe will be constructed using advanced manufacturing techniques. The primary material used will be GRCop-42, a high-strength copper chromium niobium alloy which retains properties at elevated temperatures. The alloy was specifically developed to be compatible with additive manufacturing processes. Additionally, the probe will be coated with tungsten to manage the high temperatures and plasma-material interactions.

A novel closed loop and conjugate modeling technique was developed to unify and streamline design optimization by modifying the open-source Heat Flux Engineering Analysis Toolkit (HEAT) and thermo-mechanical solvers to couple plasma heat loads to probe performance as a function of probe position within the scrape off layer. Details of this coupled optimization technique and the resultant optimal design will be presented.

To validate the manufacturing processes, thermo-mechanical performance of the probe and the closed loop conjugate optimization, test articles are being constructed which are to be tested in prototypic steady-state, transient, and cyclic loading conditions to validate the material, methods and the cooling channel design prior to placing this novel 3D printed material in the tokamak. High heat-flux testing will be performed in the HADES facility at CEA. Results of the test article manufacture, and testing will be presented.

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First wall design of a tokamak pilot plant using a Monte Carlo model for 3D heat flux deposition

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Tuesday Parallel 3b - Divertors and Plasma Facing Components II, Sala de Puerto Rico (Building W20 Room 202), June 24, 2025, 4:00 PM - 5:30 PM

We present a method for calculating the heat fluxes deposited on non-axisymmetric tokamak main wall components, allowing for a first-of-its-kind model for power handling in the tokamak far scrape-off layer (SOL). This model features strict global power conservation and is capable of calculating the finite radial plasma transport into magnetically-shadowed regions, which is significant when dealing with meter-scale shadows introduced by components such as poloidal limiters or antennas and is in contrast to millimeter-scale shadows introduced by castellations and fish-scales (which are commonly modeled already). This approach has been applied to a prototypical compact fusion reactor-class device's limiter set and allowed for an optimal set of main chamber limiters to be designed, accounting for a number of known far-SOL effects as well as limits to the wall power handling.

This model is based on DIV3D [1], which relies on a 'field line diffusion' model to mimic SOL plasma transport. In this model, a Monte Carlo ensemble of thermal macroparticles initialized at the last closed flux surface are stochastically traced along field lines (with a specified cross-field diffusivity and parallel velocity) until they intersect a PFC surface.

As a case study we apply the DIV3D model to inform the distribution of main wall poloidal limiters in an ARC-class [2] reactor device. We demonstrate that discrete protection limiters can efficiently reduce peak heat fluxes on recessed breeder wall components (below 1 MW/m2) in the presence of significant far-SOL plasma fluxes, modeled in DIV3D by scanning far-SOL diffusivity values. By varying the toroidal periodicity and radial standoff depth of the limiters, we demonstrate one of the tradeoffs that must be considered in first wall design: more limiters provide greater protection, but at the cost of reduced breeding area. We also present the impact that radial misalignments between limiters would have on local and global power handling capabilities. Finally, we show a preliminary protection limiter configuration that stays within first wall power handling limits, while also maintaining TBR>1 and overall device compactness. This case study makes use of recent upgrades to DIV3D, including the ability to set different diffusivities for near- and far-SOL populations, a model for ballooning-like thermal transport on closed field lines, numerical enhancements for axisymmetric surfaces, and postprocessing routines to apply heat fluxes to fine-scale meshes. These upgrades enable more accurate modeling of the tokamak far-SOL in geometries of interest for a fusion pilot plant.

[1] Lore, J.D., et al., IEEE Trans. Plasma Sci. 42 (2014) 539-544
[2] Kuang, A.Q., et al., Fusion Eng. Des. 137 (2018) 221-242

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Maintainability of fusion power plants

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Tuesday Parallel 3c - Operation, Maintenance, and Remote Handling; Power Management and Control, Kresge Little Theater (Building W16, downstairs), June 24, 2025, 4:00 PM - 5:30 PM

The maintainability of future fusion powerplants is probably the most impactful under-appreciated second-order design requirement facing the global effort.
Maintainability takes on a much greater significance for fusion because it is already clear that human intervention into most of these machines will be practically impossible for both safety and commercial reasons. This provides opportunities because enabling safe human access to fission facilities is expensive and drives onerous regulation. But designing automated and autonomous systems to build, operate, inspect, modify and decommission large complex facilities is beyond the current state-of-the-art. This should sound alarm bells, since we know that first-of-a-kind fusion powerplants will require the integration of many first-of-a-kind major systems that are hard to mature and test before first operation making the likelihood of early failure is high. Furthermore, engineers and scientists will propose options and upgrades. This means maintainability needs to become a first order design constraint for all power plant systems that will become activated or contaminated, and, the maintenance system, including facilities, machines, control systems and operations, need to be matured early, before finalising both machine and facility design and construction.

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The SPARC Plasma Control System software and simulation framework

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Tuesday Parallel 3c - Operation, Maintenance, and Remote Handling; Power Management and Control, Kresge Little Theater (Building W16, downstairs), June 24, 2025, 4:00 PM - 5:30 PM

The SPARC Plasma Control System (PCS) is responsible for controlling plasma parameters during pulses, and enables monitoring of critical parts of SPARC, including coils and plasma facing components. The software processes hundreds of diagnostic signals, estimates the device state, commands actuators to track operator requests, and responds to off-normal events. PCS is developed using neutrino, a lightweight software framework created by CFS for implementing real-time, high-speed, reliable code for a variety of platforms (linux, mac, embedded). The code emphasizes modularity, determinism, safety, and reproducibility and uses high speed, lock free inter-process and inter-node communication. Major algorithmic components of the PCS have been implemented, including equilibrium reconstruction, shape control, vertical control, power balance monitoring and control. A real-time heat flux and temperature model has been developed to monitor for conditions that could damage plasma facing components. An off-normal warning system has been developed to support disruption avoidance and machine protection functions by enabling transitions to soft/hard landing sequences in response to events. The neutrino framework is used for all SPARC real-time systems, including diagnostic processing like bolometer inversion and spectral line fitting. Hardware-in-the-loop (HITL) and hardware-out-of-the-loop (HOOTL) simulation frameworks, have been developed and enable connecting real-time code to both a real-time-capable control-oriented simulator (COMET), and higher fidelity simulators (MOSAIC). The simulations frameworks are used for automatic continuous integration (CI) testing, and support running parameter scans and fault scenarios.

Evaluating laser ultrasound for the in-situ inspection of plasma-facing components.

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Tuesday Parallel 3c - Operation, Maintenance, and Remote Handling; Power Management and Control, Kresge Little Theater (Building W16, downstairs), June 24, 2025, 4:00 PM - 5:30 PM

The plasma facing components (PFCs) of future fusion power plants will face a harsh and unprecedented operational environment. Their exposure to edge plasma, neutron radiation and strong electromagnetic forces poses an extreme combination of in-service loads that is expected to damage and degrade PFCs over time. Excessive PFC degradation could lead to a variety of failure accidents (e.g. an in-vacuum vessel coolant leak), with certain scenarios posing threats to nuclear safety. Whilst PFC degradation can be investigated using theoretical simulations and test rigs, accurately replicating combined PFC loads outside a fusion device is very challenging. Hence, predictions for PFC degradation and safe service lifespan in first-of-a-kind fusion power plants will carry significant uncertainty. In the face of this uncertainty, methods to directly inspect and monitor PFCs for damage in-situ during their service lifespan could help to prevent dangerous failure accidents through the early detection and characterisation of defects. Inspection data will enable PFC replacement and repair activities to be scheduled optimally, maintaining machine safety whilst avoiding over-conservatism. Additionally, the data gathered from regular in-situ inspections could improve scientific understanding of PFC ageing mechanisms and feed back into improved PFC designs. A suitable non-destructive inspection technology for this purpose must possess certain critical capabilities. Not only must the technique be capable of detecting and characterising the relevant types of PFC defect (e.g. cracks, thickness changes due to erosion & interfacial delaminations), but it must be remotely deployable in the extreme environment of an aged fusion reactor (i.e. compatible with ultra-high vacuum, radiation, magnetic fields etc.). Laser-based ultrasonic inspection has the potential to meet these requirements. By both generating and detecting ultrasound pulses in a target using laser beams, the advantages of ultrasound for through-thickness inspection can be leveraged without the need to make physical contact with the component or apply ultrasonic coupling fluid. In this work, the authors report on laboratory-scale experiments to assess the suitability and value of laser ultrasound for the in-situ inspection of PFCs. A twin-laser scanning system has been used to conduct non-contact ultrasonic inspections of a variety of PFC-relevant samples. Ultrasonic imaging has been used to detect and localise sub-surface defects in a bulk tungsten sample, as well as measure the depth of curved tungsten/copper interfaces beneath the plasma-facing surface in an ITER-like monoblock. Additionally, the detection and localisation of surface-breaking cracks in tungsten armour layers has been demonstrated using laser-excited surface elastic waves. A novel method for layer thickness measurement using thermoelastic laser ultrasound excitation has been developed and demonstrated on tungsten tiles with thicknesses of 2-10mm simulating various levels of erosion. Finally, the authors demonstrate deployment of the laser ultrasound scanning system on a robotic arm, simulating practical in-situ deployment. The results of these experiments are discussed, to assess the level of capability that has been demonstrated and identify priorities for development toward the use of laser ultrasonic inspection inside future fusion devices.

A fast double-null control concept enabling compact tokamak reactors.

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Tuesday Parallel 3c - Operation, Maintenance, and Remote Handling; Power Management and Control, Kresge Little Theater (Building W16, downstairs), June 24, 2025, 4:00 PM - 5:30 PM

We propose a novel concept for controlling the X-point fluxes of double null diverted (DND) plasmas in conjunction with – and on an equal or faster timescale to – the vertical position control. To realise a reactor relevant plasma in a compact DND design, divertor heat loads must be precisely balanced. However, these highly elongated plasmas are vertically unstable and fluctuations in vertical position result in divertor heat load imbalances.

A fully connected DND configuration minimises transport from the outboard to the inboard scrape-off layer, ensuring that a majority of the heat exhaust goes to the outboard side which has a larger surface area and divertor volume, [1][2] and makes designing the inner divertors easier. In addition to reducing costs by increasing component lifespan, maintaining a sufficiently connected DND configuration may also be a necessary requirement for H-mode access and thus for the operation of the plant at sufficiently high fusion power to produce net electricity [3].

The new control system would use fast in-vessel coils to perform isoflux control on the plasma X-points to maintain a fully connected DND configuration, even during rapid fluctuations and transient events where isoflux controllers using ex-vessel coils would fail to do so. This system could be a key enabler for ensuring sufficiently high performance and cost-effective future power plant designs.

We have numerically validated the control concept on models of MAST-U, TCV and STEP using MEQ's FGE free boundary equilibrium code [4], demonstrating sufficiently fast isoflux control with practical power supply and coil requirements. This concept will be tested with experiments on TCV. We discuss these results and the potential implementation of the concept on the STEP prototype power plant.

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[2] Osawa, R. T., et al. Nuclear Fusion 63.7 (2023): 076032.
[3] Meyer, H., et al. Plasma physics and controlled fusion 47.6 (2005): 843.
[4] Carpanese, Francesco. Development of free-boundary equilibrium and transport solvers for simulation and real-time interpretation of tokamak experiments. No. 7914. EPFL, 2021.

Modeling disruption forces and vertical stability in a novel 3D anisotropic vacuum vessel design

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

As fusion energy approaches commercialization, the scale of tokamaks is expected to increase. Grid-scale devices will likely run with higher plasma current, which can significantly increase the forces experienced by conducting reactor structures during disruption. Research devices have managed to somewhat mitigate the damage from these forces with thick conducting structures to spread out stresses. In fusion power plants however, the requirement to efficiently remove much more heat through the vacuum vessel will drive structures towards thinner dimensions. There is thus a conflict between resilience against disruption forces and efficient/economical heat removal.

One approach being explored for these competing design constraints is to engineer the vacuum vessel such that it has a weaker electromagnetic coupling to disruptions, thus reducing disruption forces and allowing thinner structures which remove heat more efficiently. In a vacuum vessel constructed from a mix of electrically conducting and insulating materials, the geometry can be designed such that induced currents are reduced and even eliminated for certain plasma modes during disruption. Here we present numerical modeling of the plasma-vacuum vessel coupling using the ThinCurr and TokaMaker codes. Disruption evolution and induced forces are modeled, and different vacuum vessel conductor geometries are considered. Implications and design constraints relating to vertical stability are also analyzed.

Technology Transfer and the Necessity and Opportunity of Collaboration

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

With increasing public and private funding into fusion, the expectations for the parallel development of spillover benefits are increasing, which can only happen through enhanced cross-sector collaboration.
Across the world, fusion research has historically been funded through public programmes, resulting in a huge breadth and depth of expertise and intellectual property within the public sector. Now there are also over forty private fusion companies with innovative ideas working on diverse approaches to delivering commercial fusion power. Investors are interested in the promise of a groundbreaking clean baseload energy source, and the supply chain to support it.
This shift in the funding environment and the expectations of the funders, including the tax-paying public, means that spillover benefits from fusion into other sectors - including near-term benefits with societal, economic or financial impact - are increasingly important for both fusion's success and its case for continued investment. A key factor in achieving both commercial fusion and spillover benefits is through fostering cross-sector collaboration to understand the opportunities and to create pull from those adjacent areas. Successful examples can be found across the breadth of fusion technological challenges. Fusion plasma research has contributed to advances in many fields from the production of integrated circuits through plasma etching to sterilization in the food industry. Experience in joining fusion components was also used to manufacture the fuel tanks of the Rosetta and Beagle 2 spacecrafts.
Areas where the fusion community can act collectively to explore adjacent sectors across different technologies and the potential opportunities that UKAEA have identified and progressed will be described.

Repair Strategies and Status of the ITER Cryostat Thermal Shield

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

The ITER Cryostat Thermal Shield (CTS), which confines the radiation heat load from the cryostat and other in-cryostat systems and keeps it from being transferred to the magnets operating at 4.5K, is actively cooled by gaseous helium running through a network of cooling tubes welded to the panels.

Stress Corrosion Cracking (SCC) was identified at the cooling tubes of the Vacuum Vessel Thermal Shield (VVTS) and further investigations and inspections on CTS panels and pipes have determined SCC was a generic issue also applicable to CTS. Similarly to the remedial actions implemented on VVTS, it has thus been decided to remedy by repairing the CTS. For most CTS, the repair strategy is however different from what has been applied to the VVTS.

Addressing the different parts of the CTS, i.e. Lower Cryostat Thermal Shield (LCTS), Support Thermal Shield (STS), Equatorial Cryostat Thermal Shield (ECTS), Upper Cryostat Thermal Shield (UCTS), this paper will focus on the various strategies to remedy the SCC issue: variation of design or repair using same configuration, in situ repair or desinstallation and repair in workshop, repair vs re-manufacture etc. The status progress and latest details of the repair activities for each CTS parts are also presented. The compliance of the repair/re-manufacture timeline with the Tokamak machine construction needs is discussed.

International collaboration to identify gaps in codes and standards required for fusion development and commercialization

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

To progress fusion development towards commercialization it is important to identify the relevant codes and standards (C&S) which apply to different fusion technologies. The International Atomic Energy Agency (IAEA) helps to foster international collaboration and coordination to identify gaps in physics, technology and regulation for the peaceful deployment of fusion energy. For a safe, reliable and robust fusion industry it is important that codes and standards are developed as the industry matures. Such codes and standards should reflect the environmental effects of fusion, such as high-energy neutron radiation and high field applications. However, in cases where components do not experience these specific fusion effects, existing industrial codes and standards may be appropriate, or new ones may need to be developed.
The IAEA is bringing the international community together to facilitate the development of a technology neutral list of C&S which will be required for the commercialization of fusion energy. In identifying the C&S required, it is possible to identify gaps in C&S which need to be developed or leveraged from existing industrial C&S. In this presentation the IAEA will introduce their dynamic C&S database, which allows for international experts to collaboratively edit and discuss C&S required for commercialization of fusion energy. Case studies and best practices will also be developed, where possible, on approaches for adopting existing codes and standards in the design, construction, and operation of fusion machines. The database is one of the first modules being developed on the FUSION CONNECT platform, which is being created by the IAEA.
In addition, the IAEA will present additional modules which will be available on the CONNECT platform and a short summary of other relevant activities being conducted at the IAEA, including the World Fusion Energy Group initiative, publishing of the World Fusion Energy OUTLOOK and Fusion Key Elements document, as well as efforts encouraging public-private partnerships.

STEP to delivery – turning technical objectives into plant requirements

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

The STEP programme has a mission to "Deliver a UK prototype fusion energy plant, targeting 2040, and a path to commercial viability of fusion." This has been further developed into a set of objectives for the programme, which for the technical delivery of the plant are defined as:

1. Deliver first plasma as close to 2040 as practicable, and the following outcomes as soon thereafter as practicable.
2. Deliver net power output of at least 100MW, sustained and stable for a representative period of time to have confidence in continuous operation, repeated a representative number of times.
3. Demonstrate at least self-sufficiency in production and processing of tritium fuel from the SPP to sustain continuous power generation operation. Ideally, generate surplus tritium to support subsequent commercial plant start up.
4. Demonstrate high grade heat output relevant to applications other than electricity generation.
5. Demonstrate a route to commercial levels of plant availability, including short maintenance interventions.

In principle these give a clear set of targets that enable a powerplant design to be developed, however it is possible to interpret these in multiple ways. The uncertainty in the design space can also lead to setting requirements that are highly unlikely to be achieved leading to further increase in design risk. Aspiration and realism are finely balanced in a first of a kind fusion energy plant. It is important to drive development of the fundamental technologies required to deliver an integrated fusion energy plant with some degree of optimism, but equally important to not set the bar for success so high that it will be unachievable. Balancing these factors allow decisions to be made on specifically what the STEP Prototype Powerplant (SPP) is going to deliver and how it will deliver; decisions that are critical to enabling an integrated powerplant design to be developed.

This paper describes how these objectives have been understood by the STEP team and how the SPP will demonstrate each of these objectives. Firstly, the overarching concept of operations (ConOps) will show how the different phases of operation, and the respective operational modes enable the performance of the SPP to be initially baselined and further optimized. The objectives themselves are then addressed, identifying what specifically is required to deliver against that objective and how this will be qualified and what opportunities there are for performance development.
Finally, the paper provides an overview of the planned work for the next tranche of the programme, where the objective is to build confidence in technologies required for the SPP concept, while increasing the design definition by improving the breadth of understanding of the whole plant.

Astral Systems’ Novel Lattice-Confinement Derived Multi-State Fusion Reactor: Architecture and its Prospective Applications

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

Neutron sources utilizing gas fusion fuel, such as deuterium (D) and/or tritium (T), are well developed and deployed for various applications globally. However, their efficiency is below 10%, and their neutron intensity output is less than 5E8 n/s DD neutrons and 1E9 n/s DT neutrons, due to the low fusion cross-section probability and engineering limitations. These limitations render them unsuitable for high-duty applications such as medical radioisotope generation, boron neutron capture therapy (BNCT) diagnostics and treatments, and fission-fusion-driven systems.
To address these challenges, Astral System has developed an innovative reactor architecture based on multi-state (MSF) and lattice confinement fusion (LCF) principles. This approach aims to overcome the existing limitations by shifting the dominant fusion environment from the plasma gas phase to the material phase. In the material phase, the fuel concentration is significantly higher than in the gas phase, resulting in a higher fusion efficiency and fusion cross-section for DD and DT fuels, consequently achieving higher neutron flux performance.
The Astral Mark I neutron reactor has been designed, fabricated, commissioned, and tested, achieving a target neutron intensity output of 1E9 n/s for DD neutrons. The Astral Mark II neutron reactor is currently in the fabrication stage, with a target neutron output of over 5E9 n/s for DD neutrons and over 5E10 n/s for DT neutrons.
The Multi-State Fusion concept, along with the performance test results of the Astral Mark I reactor, will be introduced and discussed. Additionally, the commercial implications of this new class of fusion reactor, including medical radioisotope production, BNCT, and other industrial applications, will also be explored.

Toward neutronics models for stellarator systems codes

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Stellarators, which offer advantageous properties for a fusion power plant, such as steady-state operation and improved plasma stability, have received renewed interest due to recent advances in plasma theory and computational power.
However, due to their flexibility and diversity of configurations, stellarators have a large design space that must be explored before narrowing down toward a power plant design.
Systems codes are a valuable tool for performing design exploration, as these enable rapid evaluation of different reactor designs by integrating key physics and engineering models for power plants.
One challenging aspect of stellarator systems codes are the computation of neutronics responses, since these depend on a wide range of parameters, including blanket design, reactor materials, or plasma configuration, among others.
In this work, we present a preliminary neutronics model built with machine learning from a database of stellarator configurations.
OpenMC was employed to perform the neutronics simulations on 3D stellarator models created with ParaStell and DAGMC.
This surrogate model can be used with systems codes for a rapid calculation of neutronics responses, such as tritium breeding ratio or nuclear heating.

Supply and Demand of Tungsten for a Fleet of Fusion Power Plants

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To enable the widespread adoption of nuclear fusion power plants, a reliable tungsten supply chain is essential for plasma-facing and radiation shielding components in spherical and D-shaped tokamaks. The ARIES-ST and EU-DEMO1 design points were used as the basis for neutronic modelling to evaluate tungsten consumption during 40 full-power years (fpy) at 500MWth and 2GWth fusion powers. Four materials were considered for radiation shielding: ITER Grade W, tungsten carbide (WC), tungsten boride (W2B), and WC/Co. In spherical tokamaks, the central column radiation shielding, due to its proximity to the plasma, was found to be the primary consumer of tungsten. In contrast, the EU-DEMO1 design demonstrated minimal consumption by the shield due to increased reactor volume and shielding via the breeder blanket. Over 40 fpy, the ARIES-ST reactor consumed 4,231 tonnes of tungsten at 500MWth and 29,034 tonnes at 2GWth, while EU-DEMO1 consumed 3,945 tonnes at 500MWth and 9,554 tonnes at 2GWth, with the 2GWth EU-DEMO1 model being the most material efficient design in the context of a reactor roll out model. Three tungsten supply scenarios were explored, highlighting the need for new mining resources by the mid-2040s to ensure a sustainable supply for fusion plants by 2100. If the UK or US were to operate fusion power fleets without domestic tungsten sources, their supply would likely fall drastically short without heavy investment and expansion.

ITER Electron Cyclotron Ex-Vessel System progress towards industrialization and series manufacturing

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ITER's Electron cyclotron (EC) system is required to provide up to 80 MW of microwave heating and current drive to the plasma, to mitigate plasma instabilities, assist in plasma breakdown and burn through phases, and to contribute on providing pure plasma heating. The EC system is comprised of mm-wave Launchers inside several ITER Vacuum vessel (VV) ports, which are connected to their Ex-vessel First-Confinement-System (FCS), and to the Gyrotron plant through the Transmission Lines (TL).
The EC Ex-Vessel (EW) System contains several components, mainly the corrugated CuCrZr 50mm diameter Waveguides (WG), Mitre-Bends (MB), Diamond Window Units (DWU) with the brazed diamond disk, and a complementing Isolation Valve (IV), all of those assisted by a dedicated cooling system, support structures and diagnostic instrumentation. The EW system assembly relies on bolted flanges with Double-Metallic-Seals (DMS) with vacuum monitored interspace, and detailed support systems to allow the required flexibility while minimizing transmission losses to less than 2%. The system requirements include also Primary Safety Confinement for Tritium and activated dust and Ultra-High Vacuum tightness. Moreover, the design and manufacturing shall be compliant with nuclear code RCC-MRx Ed. 2022 rules for ITER operational scenario, including 1MW mm-wave loads from Gyrotron operation, nuclear heating from plasma, electromagnetic forces, and dynamic loads from seismic events. Additionally, due to the confinement requirements DMS leak-tightness is needed, while the assembly shall accommodate several millimeters of uncertainty in the position of the VV ports and other interfacing components. The EW manufacturing requires deep drilling of components, corrugation of inner surfaces with low accessibility, turning and high precision machining of flange and gasket surfaces and cleanliness and Ultra High Vacuum (UHV) compatibility.
Fusion For Energy (F4E) is the European Procurement Agency for the design and manufacturing of all the EC EW system for ITER, to be installed in four VV Upper Ports and one Equatorial Port, accounting of more than 500 components to be delivered to the highest nuclear quality and safety standards. The design effort started from the work performed in previous phases of the project (2017-2019), which included successfully prototyped components and tests in the Test Facility Launcher Components (FALCON) laboratory. This paper focuses on detailing the main design evolution, verification analyses, qualification campaigns, and prototypes planned for the upcoming industrialization and series production of the EC Ex-Vessel System foreseen to start in 2025.

Uncertainty Quantification of Homogenization on Fusion Neutronics

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Fusion neutronics is crucial for understanding fusion device operational performance and safety. When performing neutronics analysis of a complex device such as a fusion tokamak, the small details in the device causes difficulties in meshing or geometry representation and can increase computational or analyst demand to unpractical levels. To mitigate this, homogenization of the device parts has been widely used. As an example, the complex structure of the Dual-coolant lead-lithium (DCLL) blanket, consisting of flow channels, walls, coolant, and small gaps, can be difficult to mesh or represent in a neutronics model. To mitigate this, the entire DCLL region is homogenized into a single volume with volume-fraction-scaled material composition. Additionally, the material property, such as temperature and density, distribution within a part (e.g., blanket volume) has also been traditionally assumed to be constant. This work attempts to quantify the impact of such homogenization approaches and explore methods to mitigate errors that arise from homogenization.

This is done by: 1) Developing the capability to model material and material properties explicitly, 2) comparing the differences between explicit modeling and homogenized modeling, and 3) demonstrating scaling methods to reduce the errors introduced by homogenization. This work demonstrates this on a randomly-generated channel-case geometry to explore the sensitivity of the homogenization errors with the channel volume fractions as well as the spatial distribution of the channel. Additionally, this comparison is demonstrated on a simple tokamak geometry by comparing homogenized channel geometry and explicitly-modeled channel geometry for the first wall cooling channel.

Results show that the range of error introduced by homogenization can be as high as 20% for volume-average values like tritium breeding ratio, and are sensitive to the volume split between the different materials and the distribution of the actual parts within the homogenized volume, as well as the relative cross section difference between the two materials. Results also show that the homogenization errors can be significantly reduced by scaling the tallies by the macroscopic cross section ratio between the explicit and homogenized material.

Purpose, Design, and Analysis of the NSTXU Centerstack Casing Lateral Support Shims

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The NSTXU centerstack casing surrounds the central magnet core of NSTXU. It provides the inner vacuum boundary of the plasma chamber, and supports the radially inboard plasma facing components lining the plasma chamber. The casing is supported like a cantilever, fixed at the lower flange by a connection to the TF Bundle through a G-10 ring. It is flexibly supported at the top by bellows, and insulated shims intended to react the largest part of the inventory of the disruption loads at the upper support, thereby protecting the bellows from excessive loads.

A large part of the qualification of the casing and tiles requires addressing and quantifying disruption loads. These have been estimated based on extensions of behavior of NSTX during its run period roughly 15 years ago. Structural qualification of the Upgrade used these estimates of the loading as input to calculations to demonstrate a “most probable” adequacy of NSTXU for its intended operation. The estimates are included in a requirements document to guide conservative design of all components loaded by disruptions. However, the changes made for the upgrade made the extrapolation of disruption behavior uncertain. Disruptions include axisymmetric loading from axisymmetric currents induced in the vessel shells. In addition more complex currents called Halo currents are a result of plasma poloidal currents impinging on the casing. The halo currents add another layer of uncertain behavior and net lateral loading to the casing. As a result, NSTXU is equipped with a number of instruments intended to assess the nature and severity of disruptions.

The casing shims are required for lateral support of the casing, but these have also been converted to load cells to provide another measure of the magnitude of symmetric and asymmetric loading. Some of the logic in the development of the requirements document will be presented. Electromagnetic analysis of axisymmetric and halo loading will be described. Global simulations of the casing transient dynamic response, and local analysis and testing of the shim/load cell will be included. The shim load cells and other instrumentation will be an important guide in planning NSTXU experiments beyond its next run period expected in 2026.

UKAEA CHIMERA PbLi Liquid Metal Loop: Presenting the Preliminary Engineering Design Status and Description

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The UKAEA PbLi Liquid Metal Loop is an extension to the existing CHIMERA (Combined Heating and Magnetic Research Apparatus) test rig capabilities. Located at the Fusion Technology Facility (FTF) Yorkshire, the system will operate as a standalone loop for conducting experiments or to supply lead-lithium (PbLi) to the CHIMERA vessel. This capability allows the facility to perform in-vessel component tests to allow for experiments involving PbLi in representative operating conditions, enabling qualification and verification of components and assemblies operating with liquid metals. This enables the testing of magnetohydrodynamic (MHD) effects in liquid metals and prototype fusion components to support numerical predictions and validate design concepts. This system will support the technical development of fusion reactors and the STEP programme.

This paper provides an engineering design description and details of the capabilities of the loop to support fusion engineering development.

The project required a diverse range of operating conditions to be available, volumetric flow rate control being a key design criterion. The design is targeting operation across a volumetric flow rate range of between 0.02 m3/hr and 6 m3/hr, with a resolution of control of +/-0.01 m3/hr for the lower flow rates and +/-0.3 m3/hr at higher flow rates. To achieve this large flowrate range the loop has a main loop for high flow requirements and a sub-loop for low flow testing and purification apparatus. This permits for the versatile testing of liquid metal behaviours from low flow PbLi tritium breeder blanket modules to the investigation of liquid metal cooling channels or distribution manifolds.

The loop has inline purification with magnetic and cold traps to reduce impurities. Compositional analysis of the PbLi can be conducted with a safe sampling facility which takes representative samples from the process line or storage vessel. Thought was taken in how to best get a representative sample for the system and how to then adjust back to a eutectic mix.

The loop supports a large test piece, up to 3.2m x 1.8 x 1.1m volume, with a PbLi supply capacity of up to 270 litres allowing full size breeder blanket scale testing. The test section within the loop is insulated and heated to reduce design-burden on experimenters who want to test different test pieces without needing to worry about heat loss or include insulation.

A control and data acquisition system supports safe operations and can accommodate a high number of inputs from test pieces to give detailed information for postprocessing. The control system interfaces with the CHIMERA system architecture to allow multi-physics testing of test pieces in CHIMERA with PbLi.

Due to the proximity of high-pressure water and the PbLi liquid metal in CHIMERA when testing WCLL breeder modules, safety features are included to mitigate PbLi-water interactions and avoid catastrophic consequences.

The loop preliminary design will be completed by March 2025 with detailed design and build occurring in 2025.

Shutter-Mirror Linkage Prototype Test for ITER ECE Diagnostic

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The electron cyclotron emission (ECE) diagnostic system for ITER plays a primary role in measuring the plasma core electron temperature profile and the electron temperature fluctuation with high spatial and temporal resolution. To ensure accurate measurements, the ECE system must be calibrated using an in-situ blackbody hot source capable of reaching temperatures of 700°C to 800°C. The calibration involves opening a shutter–mirror via a linkage mechanism, allowing the system to switch its view from the plasma to the calibration source. A proposed linkage design, connecting the shutter–mirror to an actuator, converts the actuator's linear motion into the rotational motion required to operate the shutter–mirror. To evaluate the feasibility and reliability of this design, a prototype linkage system is developed and tested. Initial tests include 2,000 operational cycles under ambient conditions to establish baseline forces for operating the shutter–mirror. Subsequently, the system undergoes an additional 1,500 operational cycles under vacuum conditions at elevated temperatures to simulate the ITER environment during ECE diagnostic calibration. These tests help analyze the impacts of increased friction on moving components caused by vacuum and high-temperature conditions. The prototype successfully completes 3,500 test cycles, far exceeding the approximately 200 cycles anticipated during ITER's lifetime. The results demonstrate the linkage system's robustness and reliability, providing valuable insights for optimizing the design of ECE components for ITER.

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Tritium Laboratory Karlsruhe: Overview of current tritium-based R&D efforts for fuel cycle technology

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

For over three decades, the Tritium Laboratory Karlsruhe (TLK) at KIT has been developing, operating, and testing a wide range of tritium technologies. TLK's original scope, the development of reference processes for the fuel cycle of a future fusion reactor, has been expanded to include the design and operation of tritium loops for the neutrino mass experiment KATRIN. Central to these objectives, TLK operates a closed tritium loop with an inventory of up to 40 g.

Beyond the reliable operation of a closed tritium cycle as a research infrastructure for experiments with tritium on a technical scale, TLK engages in a broad array of research efforts within the Fusion Roadmap. These include:
(i) fundamental tritium properties and tritium-material interactions due to the radioactive nature of tritium,
(ii) tritium and hydrogen analytics and accountancy,
(iii) tritium qualification of processes and scaling-up of systems to technical scale,
(iv) tritium decontamination, safety, and waste management.

This presentation offers an overview of TLK’s current research thrust towards fuel cycle technologies, focusing on research highlights in fundamental tritium properties and analytical methods.

Spatial characterization of plasma properties in a Kamaboko negative ion source

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High energy Negative ion Neutral Beam Injectors (N-NBIs) operated so far have employed filament-based negative ion sources, unlike the ITER N-NBIs, which will instead use Radio-Frequency (RF)-based negative ion sources. These two ion source types differ significantly in their plasma generation methods, geometry, and magnetic field topology. Therefore, characterizing the plasma properties of filament-based negative io sources is crucial for identifying the key factors underlying their effectiveness, with the aim of supporting the R&D activities for the ITER NBI ion source.
Since negative ion sources are based on the tandem concept, it is essential to estimate plasma properties both in the arc region, where ionization occurs, and in the extraction region, where negative ions are generated. To this end, a water-cooled movable Langmuir probe was developed at Consorzio RFX and operated at the Negative Ion Test Stand (NITS) facility in QST (Naka, Japan), specifically in a Kamaboko-like negative ion source. This study presents axial profiles of plasma properties –plasma density, electron temperature and plasma potential– measured from the arc region to the extraction region under varying experimental conditions, including source pressure and wall polarization. A comparison with spectroscopic measurements obtained under the same conditions is also provided. The experimental activities, data analysis, and interpretation are detailed, with the aim of shedding light on the physical mechanisms that may explain the performance differences observed between RF- and filament-based negative ion sources.

Turbulence modulation and re-laminarization in magnetohydrodynamic wall-bounded flows

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Fusion energy is one of the most promising pathways to large-scale, sustainable power generation. Over the past several decades, researchers have made significant advances in sustaining nuclear fusion reactions, moving us closer to a practical fusion power plant. However, major engineering challenges remain before fusion energy can be harnessed for commercial use. One such challenge is designing the blanket system, which simultaneously functions as a tritium breeding module, shields superconducting magnets from neutron flux, and serves as a heat exchanger to remove thermal energy from the fusion core.

This system operates under extreme conditions—high temperatures, intense radiation, and strong magnetic fields—so choosing an appropriate working fluid is critical. Liquid metals like PbLi have attracted attention for their high heat transfer capability and tritium breeding potential, but they also pose challenges such as high chemical reactivity and large magnetohydrodynamic (MHD) pressure drops. Molten salts like FLiBe offer lower MHD pressure drops, enabling higher flow rates and more efficient heat removal. However, these elevated flow rates can cause turbulent flow regimes that require careful study to ensure reliability and performance.

In this work, we analyze wall-bounded turbulent channel flows under an external magnetic field. We assume incompressibility and a low magnetic Reynolds number, so flow-induced variations in the magnetic field are negligible. We use direct numerical simulations (DNS) with a Fourier-spectral method in the streamwise and spanwise directions and a high-order B-spline method in the wall-normal direction. Results show that moderate Hartmann numbers (Ha) do not fully suppress turbulence but significantly reorganize it, reducing dominant turbulence length scales compared to non-MHD flows.

At higher Hartmann numbers, there is a clear transition toward re-laminarization, with diminished large-scale turbulence and more pronounced streamwise-elongated structures. Spectral analysis confirms that increasing Ha dampens large eddies while emphasizing near-wall, elongated flow features.

We also explore transient behavior by suddenly imposing a magnetic field on a fully developed non-MHD turbulent flow, revealing how flow structures and turbulent energy distribution evolve under MHD conditions. These findings improve our understanding of turbulence modulation by magnetic fields and guide the design of advanced blanket systems that enhance heat transfer and tritium management, advancing fusion energy toward commercial viability.

Accelerated Discretized Geometry: Advancing Computational Tools for Fusion Energy Simulations

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

The accelerated discretized geometry (XDG) library is designed to enhance high-performance particle transport and ray tracing operations on complex geometries. Aimed at advancing computational tools for fusion energy research and other scientific domains, XDG supports multiple mesh libraries widely used in scientific computing, namely MOAB, libMesh and MFEM. The library supports both triangle/quadrilateral surface meshes and tetrahedral/hexahedral linear element volumetric meshes, offering a versatile framework for applications ranging from radiative heat transfer to particle transport. In particular, XDG is being developed with a primary aim for use with OpenMC.

XDG achieves high computational efficiency through mixed-precision ray tracing algorithms and data structures tailored for particle tracking and geometric queries. In addition to CPUs, the project targets support for these operations on GPU by leveraging the general purpose raytracing (GPRT) toolkit as well. Several utilities for intersection checking are available in the code and algorithms for automated domain discretization have been implemented to support anticipated multiphysics applications.

This work showcases the interface, architecture, and capabilities of XDG, with a focus on its applicability to fusion neutronics simulations. We will highlight performance and validation benchmarks, demonstrate interoperability with multiple libraries, and discuss the project's near-term roadmap. By addressing the challenges of particle transport in complex geometries, XDG aims to provide support for a multitude of workflows in fusion neutronics analysis.

A Small-Scale High-Temperature Gas-Atomization Furnace for Developing Reduced Activation Alloys for Laser Powder Bed Fusion

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

Additive manufacturing of fusion reactor components through Laser Powder Bed Fusion was previously demonstrated with GRCop-42 (Cr2Nb) copper alloy powder created through gas atomization—where molten GRCop-42 was broken into a fine powder with a high-velocity stream of inert gas. Neutron activation of Niobium in the alloy produces long-lived radioactive waste; an Nb-free alloy with similar strength is sought to replace GRCop-42. Finding candidate materials requires extensive testing of various copper alloys. Commercial atomization is readily available in bulk, however testing only requires sub-gram quantities, leading to extra waste and unneeded costs. A small-batch gas atomizer with a unique nozzle design has been developed to efficiently produce grams of consistent droplets of atomized powder. This paper focuses on the atomizer's high-temperature furnace system used to melt the alloys. The furnace body is a stainless steel Reducing 4-Way Cross ConFlat Flange with two dedicated holes for the vacuum port (bottom-side) and viewport (top-side). The viewport gives visual access to the hollow carbon tube that lies in the center of the furnace body, inside which, an alloy in testing is placed. The remaining two ends of the body are connected to water-cooled copper electrodes connected to a voltage source, allowing current to flow through the copper tube to heat it. A pyrometer is pointed at the center of the tube through the viewport to collect temperature data and a PID feedback controller system sets and maintains the furnace temperature. The body is insulated with graphite felt and the exterior is cooled with a small, high-speed fan. The furnace is operated in an inert environment, brought down to a vacuum to remove air and backfilled with argon gas to reduce oxidation of carbon tube, graphite felt, and alloy. Tests concluded there is about an 80 mm distance, 40 mm on either side of the center of the furnace, where the temperature of the furnace stably remains close to the desired temperature set for the system. Tests showed furnace was able to get to a max temperature of 2362 ºC, at a power of 1816.2 Watts, before concluding tests due to the melting of a rubber O-Ring spacer on the furnace.

Prototyping High-Frequency Mirnov Coils and Real-Time Magnetic Signal Processing for SPARC

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

High-frequency measurements of magnetic fluctuations are crucial for diagnosing magnetohydrodynamic (MHD) instabilities in the SPARC tokamak [1]. These instabilities, including Alfvén eigenmodes (AEs) driven by ICRH-accelerated fast ions and DT-fusion alpha particles, occur at frequencies ranging from ~0.1-1 MHz. To address this challenge, we have prototyped a series of high-sensitivity Mirnov coils and are developing real-time signal processing systems to enable detailed analysis of these instabilities. Prototyping efforts centered on constructing coils from 2 mm-diameter Mineral-insulated Cable (MiC), exploring variations in the number of turns, conducting sheath thickness, and transmission line lengths. A drive coil was used to generate magnetic field fluctuations spanning ~0.5-1.75 MHz, producing field amplitudes of ~1×10-5 T. SPICE modeling will be used to simulate the electromagnetic and transmission line behavior of the prototyped Mirnov coils. Real-time magnetic signal processing is then explored using Field-Programmable Gate Arrays (FPGAs) to handle high-bandwidth data acquisition and analysis, aiming for latencies in the microsecond range. This system will aim to process coil signals to extract mode frequencies, mode numbers, growth rates, and amplitudes, providing immediate insights into AE activity and other MHD phenomena.

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Development of Plasma Facing, Tungsten Coated B4C Shielding for SPARC

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

The SPARC tokamak is designed for short-pulse, 10 second flattop, high fusion power - up to 140MW in operation. SPARC uses boron carbide (B4C) inside and around the vacuum vessel to reduce neutron flux to the Tokamak Hall, and nuclear heating to superconducting magnets. A portion of this B4C has direct line of sight to the plasma and is exposed to high energy, ~keV, neutral particles and photon heat loads above 100 kW/m2 during flattop. During disruption events, flash-heating can reach several 10's of MJ/m^2/s^0.5.

To avoid contamination of the vacuum vessel by boron and carbon impurities, the plasma-facing portion of B4C will be coated with tungsten. Results from design studies and testing campaigns are summarized and demonstrate a solution that will meet SPARC's challenging performance requirements, while balancing constraints on cost, schedule and the need to produce coated B4C using existing industrial processes. Multiple coating methods were explored, and physical vapor deposition was selected. Laser-based testing demonstrated 20um coatings could survive repeated flash heating up to 35MJ/m^2s^0.5. Thermomechanical analyses and ion-beam heat flux testing were used to understand material limitations, and to screen tungsten coated B4C tiles. SPARC tiles will be exposed to surface heat flux limits exceeding 100kW/m2, sustained for SPARC's 10 second flattop, thousands of times throughout the life of the machine. These tiles must be constructed to accommodate this design criteria.

Modeling and Mitigating Beam Reionization in the DIII-D Neutral Beam Injection System Using Non-Evaporable Getter (NEG) Pumps

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

Neutral Beam Injection (NBI) is widely used in major magnetic confinement fusion devices for non-inductive heating, current drive, fueling, and diagnostics. The DIII-D tokamak features eight NBI ion sources based on the U.S. Common Long Pulse Source (CLPS), with a combined output power of up to 20 MW. Over the years, modifications to the NBI system have been implemented to enhance power output, extend pulse length, and achieve advanced control over the neutral beams injected into the plasma. However, further increases in the power delivered to the plasma are currently limited by beam reionization occurring in the drift duct region of the beamline.

Beam reionization is caused by collisions between fast neutral beam particles and the background gas. Once re-ionized, the particles are influenced by the strong magnetic fields present in the drift duct, leading to impacts on the duct walls, which cause damage and reduce the injected beam power. Minimizing damage from re-ionized particles requires reducing their formation by lowering the background pressure in the drift duct region.

In this study, SolidWorks™ and MolFlow™ software were used to model the gas dynamics and generate pressure profiles along the beamline. The model was then modified to incorporate additional pumping capacity near the drift duct, and the effects on pressure profiles were analyzed. Due to the limited space and proximity to field coils, turbo molecular pumps are unsuitable for this application. As an alternative, a Non-Evaporable Getter (NEG) pump was considered. This paper also includes an assessment of the performance and potential of a NEG pump based on a benchtop experimental setup.

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Transient Analysis of Tritium Burn Efficiency in Magnetic Fusion: Exploring the Effects of Fueling and Exhaust on Performance

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Tritium Burn Efficiency (TBE) is a critical parameter for optimizing fusion performance and achieving Tritium self-sufficiency, quantifying the fraction of injected Tritium that undergoes fusion reactions in a magnetically-confined plasma. Fuel cycle analyses show that a low TBE sets demanding requirements on fuel cycle components, and results in high tritium inventories throughout the plant. This study investigates TBE under transient operational conditions, using a simplified, 0D Python model, based on particle balance equations. The model is designed using an object-oriented approach that, leveraging Python's versatility, ensures modularity and extensibility for future implementations. A system composed of a burning plasma, a scrape-off layer and a divertor region is simulated, incorporating key plasma and technological parameters, among which exhaust pumping speed, pump particle selectivity, and fueling rate. The findings aim to guide design and operational strategies to enhance fusion reactor performance and support the design of Tritium fuel-cycle.

European workforce distribution and development in the EUROfusion Consortium

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The European and global fusion ecosystems have changed considerably over the last years. ITER needs are evolving, with increasing focus on commissioning of its plant systems. At the same time, significant private investments into fusion research and growing involvement of industry create the need for a larger, more diverse, well-trained workforce.

The EUROfusion Consortium coordinates fusion research and development across its members and their affiliated entities, a total of 197 institutes including over 100 universities across 29 European countries. In 2023, EUROfusion launched a human resource survey to provide an overview of the competencies and demographics of the European fusion community. The survey allowed us to assess the current staff, in view of current and future needs. The workforce survey is complemented with an annual review of educational programmes across the 100+ European universities, as well as a review, strategy and implementation plan for the European knowledge management activities. These three overviews allow the definition of a comprehensive educational, training, knowledge management and workforce development strategy to boost and support the European fusion community targeting gaps and matching needs that fit the future needs as identified jointly with Fusion for Energy and ITER.

As recruitment and human resource policies are local to the member institutes, EUROfusion concentrates on joint commitments and coordinated actions, focusing on:

- raising awareness and attractiveness of fusion as a career path. Recruitment and retention of qualified engineers, technicians and operators are difficult in the strong job market competition. Joint efforts in 2024 successfully increased the gender and geographical diversity, and the number of successful applications to EUROfusion grants and schools. The new European Fusion Diversity Network, a community of practice will further support individual institutes in their talent recruitment and retention efforts.
- strengthen access to comprehensive education and training across Europe through the Fusion Education and Learning Hub. At present, access to fusion-focused academic degrees and courses is limited or not accessible in many countries and universities. The new EUROfusion eLearning portal offers access to recorded university courses, live online courses and training material across Europe and globally to students, newcomers and people interested in fusion research.
- contribute to the knowledge capture and transfer of institutional/European know-how in fusion science, technology and operations. The EUROfusion Knowledge Management Strategy identified 25 specific recommendations focusing on three focus areas, the ITER Engineering Design Handbook (describing ITER’s design decisions and lessons learned), making fusion education available and accessible through Fusion Education and Learning Hub, and finally operator training and communities of practice. The EUROfusion Operations Networks creates dedicated communities of practice for operators (currently on NBI and ECRH), sharing commissioning, operational and maintenance experience, as well as offering joint R&D and training opportunities for all European teams, F4E, ITER and JT-60SA.

An overview of UKAEA activities in the development of design criteria, codes and standards for fusion

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After decades of worldwide research and development including recent world records for inertial and magnetic fusion energy, fusion engineering 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 codes and standards covering intended use in light of emerging concepts, particularly for the in-vessel components.

In this work, we provide an overview of UKAEA's contributions to the development of design criteria, codes, and standards for fusion, with a focus on tokamaks, highlighting the progress made and the challenges that remain to be addressed. This includes collaboration with the IAEA to establish a database identifying existing codes and standards applicable to future fusion power plants, using the Spherical Tokamak for Energy Production (STEP) program as a trail use case, and aligning with the latest developments in the fusion regulatory framework and safety. Within EUROfusion, UKAEA is contributing 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. Within ASME BPVC Section III, Division 4, UKAEA has been actively engaged with the consortium to develop codes and standards for the construction of fusion energy devices, representing the most comprehensive set of codes and standards applicable to in-vessel components, magnets, vacuum vessels, material qualification, manufacturing, and non-destructive examination, among other areas. By integrating these efforts, UKAEA aims to establish a comprehensive pathway to address engineering challenges, develop fusion codes and standards, and ensure compliance with regulatory requirements, ultimately advancing the path toward commercial fusion energy.

Re-entrant Lyman-alpha and soft X-ray arrays for LTX-β that achieve near-complete poloidal coverage

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The Lithium Tokamak eXperiment-β (LTX-β) aims to study improved plasma performance resulting from evaporative liquid lithium coatings on its plasma facing components (PFCs) [1,2]. Previous results from the device show that progressively adding more solid lithium to the PFCs can reduce recycling coefficients to values as low as 0.5 [3]. The recycling estimates were based on a poloidal ex-vessel Lyman-alpha array that had limited coverage of the device edge [4]. To better measure particle recycling we have designed two sets of re-entrant poloidal photodiode based Lyman alpha arrays. This design introduces these sets of re-entrant, actively cooled arrays positioned near the plasma-facing surface. This system does necessitate both shuttering for the lithium environment and active cooling due to liquid lithium temperatures greater than 200°C, but by making the array re-entrant, the design achieves near complete poloidal coverage with 104 sightlines and 1,513 sightline intersections. The photodiode arrays are modular and interchangeable, allowing for both Lyman alpha and soft X-ray acquisition. Due to a low collisionality and low recycling scrape off layer (SOL), the ionization mean free paths in LTX-beta can be of the order of the SOL width. This would likely give rise to larger Lyman alpha emission volume that is not localized close to the PFCs. A re-entrant Lyman alpha array with many sightline intersections can be inverted to get a measured 2D emission profile, providing useful insight into tokamak performance with fully covered liquid lithium walls. The re-entrant array will also serve as the soft X-ray array, by swapping the Lyman-alpha filters with beryllium foils of appropriate thickness. This provides insight into the presence of tearing mode activity and their functional dependence on recycling [5]. Given the use of two sets of independent arrays, this could also allow for the simultaneous measurement of both Lyman-alpha and soft X-rays. The details of the design will be presented, along with any initial data collected from the upcoming LTX-beta campaign.

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Transforming Fusion Diagnostic Design with Probabilistic AI

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Economic fusion energy can only be delivered through reliable operation and control of a fusion plasma and the wider plant. When designing a fusion plant, traditional approaches of diagnostics or control subsystems rely on overly manual, specific design which is then iterated over time. This can often be very time consuming and expensive, and may pose significant risks for future powerplants which have lower diagnostic coverage and harsher environments. In this work we have developed a novel methodology for designing control systems for fusion devices using Bayesian experimental design.

Bayesian experimental design is a framework to assess the ability of a system to be measured and controlled, based upon Bayesian theory and probabilistic AI methods such as Gaussian Processes. The framework inherently accounts for uncertainties in diagnostics, actuators, and models. It can be used by decision makers to assess the uncertainty or risk of a design, or make automated designs, such as the viewing angle or placement of a camera. Crucially, this is done through the lens of information gain - a measure of how well a diagnostic system can differentiate between different simulated states within uncertainty.

Practically, this novel approach represents a shift in how diagnostics and fusion control systems are designed, to be more flexible, integrated, and holistic. Flexible, in that the single framework can answer many diverse questions about a design, including the uncertainty of key quantities of interest, or how sensing performance reduces under failure. Integrated, in that it determines the added information gained by combining data from sensors which may conventionally be kept separate. And holistic, in that it assesses a sensor set’s ability not to measure one value, but to distinguish between different possible states of the fusion plant as an entire system.

The benefits of this Bayesian design framework make it seamlessly compatible with digital twins—virtual counterparts of physical fusion devices. Indeed, the basic structure of a digital twin is a set of integrated diagnostic data that is compared to simulations of the real asset, to identify which simulated state is most likely and track this digital version of the asset. The proposed framework designs exactly for this, analysing the ability for a complex integrated diagnostic system to distinguish between simulated states. Though digital twins are not the only application of this framework, they represent an application that is in need of modern design tools.

To validate the proposed method, the UKAEA and digiLab have applied the Bayesian design software to challenges across fusion. This framework was used to assess how uncertainties in equilibrium reconstruction of MAST are impacted by sensor failure. Additionally, this framework was applied to automatically recommend optimal integrated designs of multiple spectroscopic diagnostics. Though this initial demonstration work shows the capabilities of Bayesian design, this tool can be applied to a wealth of other areas in the fusion, including the monitoring of heat loads, plasma stability, and tritium accountability. And as the community evolves from scientific experiments to powerplants, these modern design methods will only become more relevant.

Prototyping the neutron activation system for fusion energy measurements in SPARC

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The total energy of neutron-producing fusion reactions, like those of deuterium (D) and tritium (T), can be assessed via the nuclear activation of carefully selected, located, and measured materials. Such a neutron activation system (NAS) [Raj RSI 2024] is planned for the SPARC tokamak, currently under construction by Commonwealth Fusion Systems (CFS) in Devens, MA, and with an expected neutron rate up to 5e19 DTn/s [Creely JPP 2020]. This contribution will describe the NAS prototyping activities of a collaboration between MIT and CFS. Thin "foil" materials - such as copper, aluminum, and indium - are irradiated with both DD and DT neutron sources (up to ~1e8 n/s) at MIT, with activation-induced decay gammas measured by a combination of lanthanum-bromide, lanthanum-chloride, and high purity germanium spectrometers. The impacts of various “rabbit” capsules - materials and designs - on both foil irradiation and gamma counting are evaluated, and results are compared with activation simulations from FISPACT [Sublet NDS 2017] and OpenMC [Romano ANE 2015]. Opportunities for multi-foil-material activation and gamma spectrum forward-fitting vs unfolding are explored, along with novel materials and nuclear reactions. Topics on the nuclear-mechanical engineering and integration of such a system, as well as optimal design and operation, may also be discussed.

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Ion optics with oval apertures for TCV NBI

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The wide variety of TCV plasma scenarios and configurations require flexibility in the operation of the injection system. In parallel with continued operation of the existing heating beams with maximal energy of injected Deuterium atoms of 25-28 keV (NBI-1) and 50-52 keV (NBI-2), a new ion optical system design for intermediate particles energy of the order of 35-40 keV is considered, which can be the optimal for medium plasma densities in TCV ($4…5×10^{19}/m^3$). Development of new ion optics for the heating beam injector with particles energy up to 38 keV has been performed, and the beam simulation results are obtained. The attempt to apply the tri-electrode system with new type of oval elementary apertures was taken. This solution provides generation of ion beams with oval cross-section, which helps a better accommodation of the beams within rectangular or oval/elliptic beam ducts and minimizes beam losses, that is important for powerful NBIs with ~ 1 MW power range. The concept combines advantages of ion optics with circular apertures (high strength and geometric stability of electrodes) together with those of ion optics with slits (reduced beam divergence in one direction, and increased beam current due to a combination of bigger density and higher grid transparency). The new design feasibility was checked by a comparison with the existing circular and slit conceptions for TCV using IBSimu code for beams generation simulations. Numerical simulations of new oval beamlet shape demonstrate low divergency, reducing beam losses in the duct, which may allow the beam pulse time increase for covering of full TCV discharge with reduced risk of beam blocking or contamination of the fusion plasmas by the overheated beam duct particles sputtering. Manufacturability assessment of full-aperture grids with oval beamlets has been done in SPC design group. The new ion optical system is supposed to upgrade NBI-1 at TCV. Beams generated by oval elementary cells can be applied in fusion devices for plasma heating and diagnostics, as well as in high-energy experimental and industrial accelerators, where the beam ducts dimensions are limited by size and have a moderate aspect ratio.

Self-radiometric calibration solution for tokamak's optical diagnostics

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The radiometric calibration of optical diagnostics used in fusion reactors is critical for monitoring and studying reactions.
Optical systems installed in particularly harsh environments will likely see their optical throughout degrades over time due to optics damages. A shutter and a cleaning system can be implemented to protect optics but won't fully prevent throughout losses. Consequently, the transmission of the system needs periodic recalibration.
It's not possible to permanently install a calibrated light source in front of the system, so one solution is to place a retroreflector behind the shutter to do the calibration remotely. Various retroreflector-based solutions have been proposed, but since they rely on knowing the reflectivity on the material, they are not immune against environmental aggressions. Therefore, the calibration will drift over time.
Many variants of design have been proposed in order to calibrate the retroreflector itself but without guaranteeing the radiometric calibration.
We propose an innovative solution that offers true radiometric calibration based on dual-shape retroreflector. It combines two types of mirrors with different effects. The first returns the light like a corner cube after three reflections while the seconds induce a diffusing effect with a single reflection. The difference in reflection number associated with a different optical conjugation allows the measurement of the intrinsic reflectivity of the material of the retroreflector as well as the transmission of the optical system. Such reflector has been implemented in the design of the CXRS Core diagnostic for F4E/ITER. It has been demonstrated experimentally by Bertin that both signals can be separated and detected with enough SNR to get a self-calibration of the reflector and thus a true calibration of the transmission optical path.
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Tritium Collection and Measurement in Small-Scale Molten Salt Breeding Experiments

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The Liquid Immersion Blanket (LIB) offers a multifunctional solution for cooling, tritium breeding, and radiation shielding in fusion devices. It consists of a large molten salt tank surrounding the vacuum vessel housing the core. Developing this concept requires a thorough understanding of tritium transport properties and chemical speciation in molten salts to inform power-plant-scale LIB designs. Experiments spanning various scales are necessary: setups with 0.1 to 1 L of breeder salt focus on tritium chemistry, permeation, and transport in stationary salt, while systems with 100 L or more address flowing salt transport and demonstrate higher Tritium Breeding Ratios (TBR). Effective tritium collection and measurement techniques are critical to fully capture bred tritium, analyze its chemical speciation, and monitor temporal release profiles.

To address these needs, we developed a tritium capture and measurement system for the small-scale experiments of the LIBRA project. In the BABY tests molten salts (FLiBe and ClLiF) in volumes of 0.1 to 1 L are irradiated with 14.1 MeV fusion neutrons. The salt is contained in a crucible under flowing gas to sweep tritium released from its free surface. A secondary vessel surrounds the crucible to trap tritium permeating through its walls, capturing it with flowing gas. The two gas streams are processed through separate sets of bubblers equipped with oxidizing furnaces, enabling differentiation of tritium chemical species. Bubblers are sampled periodically to monitor the temporal evolution of tritium release, and sample activities are measured using liquid scintillation counting (LSC).

This system detects tritium activities as low as 0.2 Bq per sample. Precise quantification of sample activity and total tritium release enables accurate TBR measurement. Temporal resolution allows for robust benchmarking of tritium transport models, while the oxidizer furnace facilitates the study of factors affecting tritium speciation. Key considerations in this procedure include LSC quench calibration, optimization of the sampling techniques, and addressing potential chemical contaminants. The insights gained from the BABY experiments can inform the design of tritium collection and measurement systems for other small-scale breeding tests.

Reactor relevant ECE measurements from a Michelson interferometer; lessons from DIII-D

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Measurements of electron cyclotron emission (ECE) will be critical for monitoring and controlling the high temperature plasmas in next step fusion devices like ITER and SPARC. A Michelson interferometer (also sometimes called a Fourier Transform Spectrometer, FTS) instrument covers a broad spectral range of microwave frequencies (typically 50-1000 GHz) and hence is able to measure multiple ECE harmonics. This multi-harmonic capability not only yields local electron temperature (Te) values across the plasma radius, but can also provide information on the electron distribution function via model-based comparison. The DIII-D Michelson interferometer has 25 years of archived data from experiments, many of which approach reactor conditions of high Te and high normalized beta. A review of the extensive data set, and the setup, calibration and operation of the instrument helps guide design and implementation on future devices.

The DIII-D Michelson employed approximately 20 m of low loss corrugated waveguide transmission line (-4 dB loss) to be remotely stationed from the tokamak. Nevertheless, the instrument's data was affected by both stray microwave power from gyrotrons and electronic spikes due to neutrons. The stray microwaves were mitigated by shielding with mw absorber plus the use of a dichroic plate high-pass filter. Software was developed to automatically remove neutron spikes from the signals. Other impacts to the data were from transients in the plasma, e.g., strong reconnection events and the ECE bursting phenomenon. Again, many of these effects were able to be handled by specially designed signal post-processing of the interferograms and overall good ECE spectra covering most of the plasma pulse were obtained. These methods are applicable to combat similar disturbances in the raw data in next-step devices.

Calibration of the Michelson interferometer was carried out between every major operations campaign on DIII-D. Due to the instrument's cryogenically cooled, single detector configuration, the calibration was found to be stable for time periods of a year or more. Accurate and stable calibration is important for the Michelson measurements in order to carefully compare emission at different electron cyclotron harmonics, which is key to ascertaining deviations from a Maxwellian distribution. These deviations are believed to be the cause of the Thomson scattering / ECE discrepancy of measured Te that have been seen in high Te discharges typical of what is expected in reactor level machines [1]. This Te discrepancy is a major concern, but a properly engineered and configured Michelson on future tokamaks will help to address the uncertainties in the measurement.

[1] Fontana, et al, Phys. Plasmas 30, 122503 (2023)

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Fusion Supply Chain Development from the UK STEP Programme – Enriched Lithium

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The mission of the UK's Spherical Tokamak for Energy Production (STEP) programme is to deliver a prototype fusion energy plant, targeting 2040, and a path to commercial viability of fusion. The path to commercial viability of fusion requires the stimulation of a broad base of industrial capability that can compete globally to deploy, operate and service fusion energy plants around the world.

This talk will summarise how STEP is enabling industry to rise to the challenges of a plant that must simultaneously deliver net energy, sustain a Deuterium-Tritium fuel cycle and a demonstrate viable route to plant maintenance.
Through an innovative delivery model that creates an integrated team of fusion, engineering and construction partners, the solutions developed for the STEP Prototype can then be taken to market, including the emerging global fusion sector and adjacent high-value sectors. This focus on long, medium and near-term economic benefit supports the case for investment in fusion technology development.

Enriched Lithium will be presented as a case study of the strategic approach to supply chain development in action. Isotopically tailored Lithium is a critical material for a wide variety of DT fusion energy plant concepts, representing the majority under development globally. But a reliable supply route that can deliver the quantities demanded by fusion energy demonstrators and commercial fusion plants thereafter, does not currently exist. The STEP Prototype's Tritium Breeder system creates a visible demand signal that can stimulate a supply chain that can go on to support other fusion energy projects, as well as complimentary demand in adjacent sectors. The talk will provide an update on UKAEA efforts to establish an enriched Lithium supply route, built on scalable and sustainable production methods. It will also identify outstanding challenges and potential enablers that support the fusion sector's ambition to provide globally available and competitive energy.

Qualification of the ITER magnetic diagnostics instrumentation: sensors and integrators

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Like in any magnetic fusion device, the ITER magnetic diagnostics play a key role for machine protection, plasma control, and plasma physics analysis. In particular, they ensure the measurements of plasma current, edge poloidal flux allowing the calculation of its shape and position that are essential inputs for the operation and control of plasma discharges. The quality of the measurements relies on the whole measurement chain, from the sensors, up to the electronics consisting of integrators for inductive sensors. On ITER tokamak, 450 pick-up coils (inductive magnetic sensors) will be installed on the inner skin of the vacuum vessel that constitute the main magnetic systems to recover the plasma magnetic parameters. Each of them will be connected to a digital integrator through ~150 m cables. This set of 450 in-vessel sensors consists of 3 types of equilibrium pick-up coils devoted to the measurement of the local magnetic field in the direction of the sensor axis and other 2 types of pick-up coils measuring the high frequency magnetic field.

This contribution presents both the qualification of the ITER in-vessel sensors installed in their housing platform and integrators. It discusses the qualification of the 450 in-vessel sensors in terms of effective area measurement, polarity and frequency response: amplitude and phase shift from 20 Hz up to 2 MHz corresponding to 5 orders of magnitude. A specific and fully automatized test-bed dealing with such a large range of frequency was developed for this application. In order to improve the accuracy of the results for each sensor, 10 measurements were performed per frequency and mean value and standard deviation are processed. The effective area measurements of all sensors show a discrepancy of the order of 0.5% indicating that the sensor manufacturing was extremely reproducible. The response of the high frequency sensors shows that they are reliable over the whole frequency range.

In parallel, integrators were developed for ITER magnetic diagnostics. They were successfully tested in laboratory with a quiet electromagnetic environment. In order to push forward the test and qualify these integrators in real tokamak environment and operation conditions, one integrator was connected to a magnetic sensor installed in the vacuum vessel of the WEST tokamak through a ~60 m long cable. Data were recorded for each plasma discharge during the WEST experimental campaigns. Thus, the qualification was completed against various plasma conditions, including disruptions, long pulse operation (514 s), etc. In addition, the results were compared to a WEST integrator connected to a sensors located at the same poloidal location but a different toroidal position. The results show that the drifts of the ITER integrator remain well below the specification (500 µV.s/hour). The plasma disruptions can be accurately monitored. Nevertheless, a noise at a frequency of 300 Hz was identified. Further tests should be performed to continue the qualification.

TRIOMA: a new Open-Source tool to improve and simplify the design of outer fuel cycles

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Efficient tritium breeding and efficient tritium extraction are two fundamental steps to achieving tritium self-sufficiency, which is mandatory to operate a fusion power plant (FPP). High TBR and reduced tritium losses enhance the economic competitiveness of FPPs and reduce the tritium doubling time. This work focuses on the Outer Fuel Cycle (breeding blanket, tritium extractors and in some design heat exchangers) of liquid breeders, such as FLiBe which is foreseen in the Liquid Immersion Blanket (LIB) design of ARC-class reactors, and PbLi foreseen in DEMO Water-Cooled Lithium-Lead (WCLL) blanket. Efficient tritium extraction systems reduce tritium leakages and the tritium inventory in the whole OFC which is beneficial to respect safety constraints. Because of these reasons, it is crucial to design a performant fuel cycle, starting from the OFC.

The open-source TRItium Object-oriented and Modular Analysis (TRIOMA) code simplifies OFC analysis for FPPs by offering a simple, python object-oriented framework. Its 0-dimensional analytical formulation supports detailed yet fast evaluations of integral parameters, including tritium inventory estimation and fluxes, under steady-state conditions for common outer fuel cycle components, enabling rapid performance estimates for OFC configurations and experimental circuits.

The code supports heat exchangers, permeation against vacuum extractors, reverse permeators, packed bed extraction columns, and breeding blankets for molten salt and liquid metal breeders, with the possibility to account for turbulence enhancers and the outer tritium partial pressure. Because the performance of each component of the OFC is tightly coupled with the interaction with the others, TRIOMA can analyze the circuit comprehensive of all components, thanks to its flexible modular structure.

Applications include preliminary OFC design, feasibility and sensitivity studies, start-up transients, pulsed operations, and integration with other open-source tritium transport tools. By simplifying complex OFC analyses, TRIOMA empowers researchers and engineers to make informed decisions during early design stages and visualize bottlenecks and challenges for achieving tritium self-sufficiency.

Tritium extraction – innovations and models for PbLi and FLiBe blankets

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The role of the tritium extraction system is to transfer efficiently and with minimal delay the tritium produced in the breeder blanket to the overall fuel cycle, it is crucial for ensuring tritium self-sufficiency and the commercial viability of DT fusion power plants. The focus of this work is on tritium extraction from either of two leading candidate fluid materials for tritium breeding, lead lithium alloy (PbLi) and lithium fluoride-lithium beryllide (FLiBe). Both have low tritium solubility, and despite differences in solvation and tritium transport mechanisms, similar extraction methods can be applied for both materials.

This presentation will unveil recent results and ongoing R&D at the UKAEA, focusing on innovations in tritium extraction systems for PbLi and FLiBe. We introduce a novel, patented (WO/2024/042311) concept: the Innovative Integrated Tritium Extraction and Recovery System (IITERS). IITERS synergistically integrates tritium extraction across a solid membrane with tritium transfer to a second fluid (e.g., He bubbles). The use of two-phase flow addresses mixing challenges at low Reynolds numbers and adds an additional extraction route, this leads to achieving high extraction efficiency in a compact design. Its horizontal flow orientation reduces compression costs compared to conventional packed bed gas-liquid contactors. Furthermore, advances and developments in similar tritium extraction technologies (e.g. manufacturing R&D or modifications for process intensification) can be directly applied to IITERS.

Key comparisons will be made between IITERS, permeators against vacuum (PAVs), and gas-liquid contactors (GLCs) in terms of extraction efficiency, plant footprint, ancillary systems, and tritium inventory. These models benefit from work done at UKAEA on the tritium extraction systems within EUROfusion and UK programmes. The comparisons include sensitivity analyses to account for uncertainties in material and transport properties.

The capabilities developed and de-risking activities thus far for modelling, simulating, and operating GLCs, and PAVs have paved the way for IITERS to soon become the reference technology for tritium extraction in PbLi and FLiBe-based blankets.

Developing "fusionics", an interdisciplinary capability bringing together control theory and software engineering to develop embedded systems platforms fit for control and protection of fusion power plants.

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Real-time control and protection systems for plasma control in Tokamaks are typically complex, strongly coupled, distributed systems. The engineering workflow requires a strong science basis anchored in high fidelity modelling and simulation. First principles models are too computationally heavy to include within control systems, so techniques to reduce the dimensionality to an appropriate scale must be applied. Control theory must be used to design algorithms which are robust and resilient. State estimation is a significant challenge. It requires close cooperation between diagnostic physics experts to bring sensor and measurement processes to a level of performance that is fit for purpose to integrate in the overall control and protection systems.

Implementation currently requires deep expertise in electronics and programmable systems. In the absence of a commercial market for plasma control grade components, teams must adapt COTS equipment. Customisation can be applied at several levels. To reduce cost and risk, maximal use of vendor features is best practice, but comes with the requirement to qualify and commission each integrated system. The integration team need to have the skills which enable them to deeply understand the internal operation of these devices to do this effectively.

Explaining this narrative effectively to senior stakeholders who own the risk and investment decisions is challenging. Effective control and protection systems are invisible when working well. They are highly intangible, and practitioners must compete against project components which have more obvious costs and benefits.

Explaining this narrative effectively to early career scientists and engineers is also challenging. The domain provides many interesting opportunities to learn and grow professionally. However, the skillset developed is not well understood, and we must compete with more mature sectors where the opportunities are clearly defined.

Explaining this narrative effectively to the supply chain is also challenging. Writing a strong procurement specification for fusion control systems must adequately distinguish the technical problems, the multi-physics and multi-engineering skill required, and the potential benefits to be derived in building capability and capacity during this emerging industrialisation phase of fusion power.

Feynman observed that "notations are powerful; invent them". To be able to better reason and communicate, we create words to encapsulate new concepts. We propose that "fusionics" could be to fusion what avionics is to aerospace. Fusionics distils the essence of the interdisciplinary challenges set out above. Fusionics practitioners will bring electronics, software, modelling and control theory together, along with a strong understanding of how they can be used effectively to deliver commercial fusion power plants.
A fusionics supply chain will create a market of interoperable components, built to standards that allow plasma controls systems to be procured as simply as industrial automation solutions are today.

Early stage fusionics projects have been funded as part of the UK Fusion Futures program. Details of progress to date and the vision for the future roadmap of this new sector will be shared in this talk.

Prototype Design of Fast Control on the Vertical Displacement Based on FPGA Device

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

The national project of experimental advanced superconducting tokamak (EAST) is an important part of the fusion development stratagem of China, which has fully superconducting tokamak with a non-circle cross-section of the vacuum vessel and the active cooling plasma-facing components. The plasma equilibrium configuration of the elongated interface has a naturally axisymmetric vertical instability, which is one of the main mechanisms causing major disruptions. So, the plasma vertical displacement control system plays a role in ensuring the safe and stable operation of the EAST experiment and improving the performance of plasma parameters performance. The response speed of the vertical displacement control system has high requirements for time. The current system's response delay mainly comes from the response delay of the fast control power supply, the intersystem communication delay, and the waiting delay caused by the control system response cycle. Since the response delay of the fast control power supply itself is fixed, reducing the communication delay between systems and the response delay of the control system is the focus of this work. An FPGA device was developed. By using the new FPGA devices, on the one hand, the sampling rate of electromagnetic measurement signals is increased. At the same time, the vertical displacement function is separated from the plasma control system (PCS) and implemented by means of FPGA code.
For the rapid development and testing of the vertical displacement code, tests are carried out on the NI cRIO device using LabVIEW FPGA. To cut costs, the project team undertakes the independent development of the FPGA hardware board and transplants the debugged LabVIEW FPGA code. Currently, the test results based on LabVIEW FPGA are in accordance with the calculation results of PCS, and the delay response is less than the original 500 µs. The new vertical displacement function, based on the newly developed FPGA hardware control system, is on the verge of completing optimization and testing.

Progress on Preparation for Integration and Commissioning of Master Control System of ITER AC/DC Converter

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

The Master Control System (MCS) is to ensure proper operation and control all individual components consisting ITER Coil Power Supply System, which delivered from several procurement arrangements. MCS is under preparation for integration and commissioning after installation at ITER site. MCS acts an interface between ITER central system and components of the plant and local systems. It is important to integrate and test all the instrumentation and control (I&C) functions if the system operates properly. MCS should ensure stable operation of state machine and integrity of communication of commands from the plasma control system for AC/DC converter operation. After the commissioning of the system itself, it goes to the next step for integrated commissioning with ITER central system. Implementation of Master Control System is described and the Preparation process and progress for the integration of I&C system of MCS with its interfacing system is presented.

UPDATED HYDRAULIC DESIGN AND ANALYSES OF WCLL MANIFOLD

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

The ENEA Brasimone Research Center together with Italian Universities is actively supporting a research activity inside the EUROfusion consortium dedicated to the investigation of the water and lithium-lead technologies applied to the DEMO Breeding Blanket (BB) and Balance of Plant (BoP) systems.
In particular, many efforts are devoted to the design and analyses of the WCLL BB manifold systems, the current in-Vacuum Vessel coolant systems consists of inlet and outlet feeding pipes, as well as the distributing manifolds. The former are in charge of either routing the cold coolant to the different BB segments (inlet pipes) or collecting the hot water to the steam generator (outlet pipes).
In the framework of the strategy of providing a robust design to the components of the Water-Cooled Lithium Lead breeding blanket for the EU-DEMO, the analytical and numerical assessment of the previously designed PMU manifold is envisaged.
The present paper aims to present the new WCLL BZ manifold design. A general description of the manifold geometry is given and a comparison with the former one is discussed. Additionally, the design analytical calculations are presented, followed by CFD numerical results for the hydraulic optimization of the water flow inside the different poloidal BUs.
At the end, a general description of the manifold test section to be tested at ENEA Brasimone Research Center is given and the scaling rationale from the real scale component is discussed. A description of the instrumentation foreseen is presented, together with the illustration of a preliminary experimental test matrix.

A Multi-State Fusion Reactor Array Control System Design and Implementation for Automated and Modular Neutron Generation Systems

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

"Multi-State" Fusion (MSF) reactors incorporating Lattice Confinement Fusion (LCF) are an emerging novel type of neutron generator which use arrays of reactors to achieve neutron production rates >10^12 n/s or 10^10 n/cm2/s from DT reactions. Using an array of small reactors simplifies typical fusion engineering challenges encountered in plasma-facing materials and magnetic plasma confinement. It removes the risk of catastrophic system failure, as a single MSF reactor failure only reduces total system output fractionally before it is replaced. MSF reactor arrays can be configured in different shapes to allow flexible irradiation patterns, this is useful for a variety of applications such as medical isotope manufacturing, inspection systems and fission initiation, however for a single operator to control large arrays of reactors accurately and safely poses engineering challenges in the hardware and control system.

To address the MSF reactor array control requirements a new control system has been designed that automates all control and monitoring elements for the operator and exploits modular engineering techniques for system scalability.

A graphical software interface has been created that can be reconfigured on-the-fly by operators to represent the location of the reactors, neutron detectors and irradiation targets to setup the system control as intuitively as possible. Automated array subset control systems allow an operator to run banks of many 10's of reactors from an industrial real-time controller via the user interface. Operators can define desired neutron flux levels at irradiation target detectors and the reactors they wish to activate, the control system then automates power handling algorithms and PID control loops to perform the irradiation. The real-time controller software architecture is written using ladder logic for safety system interface tasks and using object orientated programming to provide best efficiency for controlling large arrays of reactors that need subtle configuration differences within the array to meet the irradiation application requirements.

The graphical interface software communicates with the real-time industrial controller using industrial ethernet protocols to configure and monitor reactors. The control system is designed with a modular distributed IO architecture to simplify manufacturing and installation, each reactor has its own identical electrical high voltage supply, fuel control system hardware and is fitted with industrial remote IO in each cabinet, each modular node only requires a 3-phase power cable and ethernet cable to link them into the control system, all distributed reactors IO are controlled from the single real-time controller at the operator cabinet.

Sample irradiation neutron detectors' data monitoring and logging is integrated into the software, communicating with neutron detectors over serial connections. In medical isotope manufacturing, traceability of a samples irradiation requires a report generation system to provide evidence that irradiation was performed correctly. To fulfil this requirement the control system also has a modular data logging system to capture all aspects of the system performance, this data must then automatically generate customised reports for each irradiated sample that confirm the full functionality of the reactor system.

Influence of Laguerre Gaussian Laser dynamics on Terahertz generation inside inhomogeneous plasma

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

The research presented in this paper offers a theoretical framework concerning the self focusing and terahertz generation of LG laser beams within a non-homogeneous plasma medium. Upon incidence of the laser beam onto the plasma medium, a refractive index gradient is generated that causes the wavefront to collapse and produce a sharp focal point. In this study, we have employed the moment theory approach to derive the differential equations pertinent to self-focusing and terahertz generation. The analysis reveals that the behavior of the laser beam and terahertz radiation is significantly influenced by the parameters of both the laser beam and plasma, including laser intensity and plasma density, among others. Various modes of the laser beam exhibit distinct impacts on self-focusing and the efficiency of radiation emissions. It has been observed that the (TEM03) mode yields the highest Terahertz Generation efficiency

Modelling tritium extraction in gas-liquid contactors: effects of geometry, materials properties, and operational parameters

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

We present a novel model of tritium transport in gas-liquid contactors (GLCs) designed for tritium extraction from liquid breeder materials such as eutectic lead-lithium alloy (PbLi) and molten salts, e.g. fluoride-lithium-beryllium (FLiBe). Initial development of the model has focussed on PbLi. The purposes of the model are: to aid the design of DT fusion power plant (FPP) extraction systems at the interface between the breeder blanket and fuelling systems; to support analysis of experimental data from prototype extraction systems; and to help prioritisation of future research and development activities for FPP breeder blanket ancillary systems. The model is based upon analysis of mass transfer of tritium from a liquid (PbLi) to a carrier gas (helium) in a steady state stripping column containing a solid packing material. It incorporates Sieverts's law for the solubility of diatomic gases in solid and liquid metals via dissociation into individual atoms. The application of the model to a vertical columnar tritium extraction unit (TEU) with appropriate values for its geometry, materials properties and process variables is implemented in MATLAB. This enables quick numerical solution of the model for each set of input values providing outputs including the extraction efficiency of the column and tritium inventory in each of the solid, liquid, and gaseous phases inside it. A rudimentary sensitivity analysis of model inputs gives an initial indication of the key design and operation parameters which would enable optimisation of extraction performance. Furthermore, estimates are also made of the amount of tritium which will be lost to the extraction process either by permeation completely through the column wall or by retention in the column wall as an inventory of mobile diffusing atoms or more permanently retained in traps such as crystallographic defects. Finally, aspects of the model are extended to consider behaviour during transient periods of operation, rather than purely steady state. Initially, an assessment is made of the characteristic times associated with some of the transport mechanisms, such as diffusion in the solid packing material and permeation through the column wall. Beyond this, transient features such as a reduced effective diffusivity because of trapping in the column wall and the effects of temperature gradients across it may be provided by a simple time-evolving finite element model of the extraction process.

Modelling of Efficient Neutral Beams for Fusion Neutron Sources

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

A compact and cost-effective tokamak facility for testing fusion nuclear technology, components and materials called volumetric neutron source (VNS) was proposed in the late 1990s. The mission of compact intense fusion neutron sources (FNS) has got much in common with that of VNS. The level of neutron yield (10^18 – 10^19 n/s) makes FNS a cost-effective solution among neutron generators. They can be invaluable for many branches of technology and research, including fusion engineering, nuclear fuel breeding and waste management, material science and neutron scattering. FNS are believed to offer a shortcut to fusion applications.

Low level of fusion power (Q value) combined with continuous neutron output, implies steady-state plasma operation driven by auxiliary power. A non-inductive current-drive scenario which can sustain steady-state operation is essential for FNS and leads to a high (60-99%) contribution of the supra-thermal ion component to FNS fusion power.

Neutral beam injection (NBI) will be the main source of energetic particles for plasma fueling, heating, torque, and non-inductive current drive, with efficiency confirmed theoretically and experimentally. The NBI scheme and geometry can be optimized not only for maximum CD efficiency and neutron generation, but also for required flexible shaping: on-axis/of-axis peaked, hollow, or almost uniform across plasma.

An integrated numerical model is used to explore the expected scenarios of FNS plasma, and for NBI performance optimization. The model incorporates the detailed 3D structure of the source beam, tracks the beam species along the injector beamline, and follows the fast ions until their thermalization in tokamak plasma. The combined approach allows efficient study of plasma shaping effects in a wide range of aspect ratios - from conventional to spherical tokamaks. The results show NBI can be a plausible solution for driving plasma operation, neutron generation and profile control, when beam-plasma parameters and geometry are appropriately matched.

Tritium breeding capabilities & progress of (UK) LIBRA Project at the University of Bristol

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

The (UK) LIBRA project, based at the University of Bristol, focuses on studying tritium breeding for deuterium-tritium (DT) fusion reactors to address the global tritium shortage. This research explores the use of 6LiD as a tritium breeder material, fabricating solid pellets of various sizes and densities under inert gas to prevent contamination. These pellets will be tested under quasi-operational conditions on a dedicated breeding blanket testing platform (BBTP) developed by the University of Bristol team.
The BBTP consists of a gas management system (GMS) and a breeder blanket module (BBM). The GMS has been developed to manage the purging gas flow through the breeder material during neutron irradiation. Flow, pressure, and gas composition can be modified in a real-time within the breeder module. Currently undergoing calibration and testing, the GMS will be integrated with the BBM for neutron irradiation tests later this year.
The BBM is a modular structure that houses the breeder material and moderates the neutrons to increase the breeding efficiency of the system. It's composed of neutron moderators, reflectors and heaters to control the breeder temperature during irradiation.
A neutron irradiation bunker, designed to host the neutron source and breeding blanket experiments, has completed the simulation phase and is now under fabrication, with completion expected this summer. Simulation results and optimization insights for the bunker design will be discussed.
The BBTP, incorporating neutron conditioning materials, a 6LiD pellet vessel, and the GMS, is finalized and being fabricated. An overview of the breeder material design, tritium breeding ratio (TBR) simulations under varying conditions, and initial experimental results will be presented, highlighting key progress in tritium breeding advancements for fusion reactors.

Fusion Twin Platform: An Innovative Tool for Fusion Research and Education

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Tuesday Posters 3, Lobdell (Building W20 Room 208), June 24, 2025, 4:00 PM - 5:30 PM

The Fusion Twin Platform (FTP), recently launched at https://fusiontwin.io/, is a free web-based tool designed to democratize access to advanced tokamak simulations, enable collaborative research in fusion science, and enhance plasma physics and fusion engineering education. FTP allows researchers, educators, and students to use pre-built digital replicas of tokamaks, enabling precise simulations, exploration of machine learning models, visualization of plasma dynamics, and flexible data management. By leveraging NSFsim, a free boundary equilibrium and transport solver, FTP supports fast customizable simulations and discharge scenario development.

FTP is a cornerstone of Next Step Fusion’s mission to remove barriers to entry in fusion research, education, and collaboration. The platform supports diverse tokamak configurations, providing access to essential datasets and machine geometry while ensuring user privacy and proprietary data security. Transformative for both research and education, FTP provides tokamak simulations by offering a suite of powerful tools, including machine learning model integration, advanced visualization capabilities, and collaborative functionalities. Fully web-based, FTP requires no additional software or hardware to run, making it accessible to users worldwide. Researchers and educators can leverage these resources to conduct fusion experiments, optimize control strategies, and engage students with hands-on, interactive learning experiences, all within a secure and accessible digital environment.
Key Features of FTP:
* Fast and Precise Simulations: FTP offers tools for customizing magnetic equilibrium simulations, developing and optimizing new discharge scenarios, and evaluating plasma stability across various operational regimes with exceptional accuracy for tokamaks such as DIII-D, ISTTOK, SMART, and others.
* Customizable Visualization and Analysis: FTP provides a fully integrated environment where users can plot, visualize, and analyze data, whether uploaded or generated on the platform.
* Comprehensive Data Management: FTP enables seamless access to fusion datasets, allowing users to upload their data for analysis or download platform-generated outputs.
* Collaborative Workspace Tools: FTP includes robust collaborative features, enabling users to share workspace data with team members, create public links for wide sharing or publication, and maintain a shared context among collaborators.
* Integrated JupyterHub Environment: FTP offers a built-in JupyterHub environment with Python notebooks and advanced extensions, such as an HDF5 viewer.
* ML Demonstration and Integration: FTP serves as a showcase and integration point for machine learning tools developed by Next Step Fusion, such as a plasma boundary reconstruction ML model trained on the DIII-D experimental dataset.

Besides being a powerful tool for fusion research, the Fusion Twin Platform (FTP) also has the potential to become an exceptional educational tool. It provides educators, students, and professionals transitioning from other fields to fusion with access to realistic tokamak simulations and interactive tools that bring fusion concepts to life. FTP enables hands-on learning experiences by allowing users to explore plasma dynamics, test machine learning models, and simulate real-world discharge scenarios within a secure, user-friendly environment. Furthermore, its collaborative features, such as shared workspaces and public link generation, foster teamwork among students and professionals while encouraging interaction between institutions. By integrating cutting-edge technology with practical educational applications, FTP bridges the gap between theoretical knowledge and practical understanding.

Values of ITER and Progress of Construction

Kamada Y[1]
[1]Iter Organization

Wednesday Plenary - Yutaka Kamada, Kresge Main Theater (Building W16, upstairs), June 25, 2025, 9:00 AM - 10:00 AM

This paper discusses the values of ITER in the current accelerated fusion development worldwide: ITER is the first-of-a-kind (FOAK) industrial-scale DT fusion device far exceeding the dimensions and parameters achieved in present devices. The role of the ITER project is to move the ground of fusion development from science to industry. ITER should achieve extended burn with the fusion gain Q ≥ 10, and should demonstrate the availability and integration of technologies essential for fusion reactors.
Regarding the availability of key fusion technology, there has been great progress by the seven Parties. For example, manufacture of all 19 Toroidal Field (TF) coils, all 6 Poloidal Field coils, and 4 (out of 7) Central Solenoid modules has been completed. It can be concluded that a global supply chain and mass production of ITER-grade superconducting magnets has been established across the world. The prototype of the tungsten divertor that receives a huge amount of heat load has been completed, and establishment of mass production of W mono blocks and plasma facing unit has become practical.
Regarding construction, repair work of Vacuum Vessels (VVs), having geometric non-conformities in the field bevel joint, has started based on successful R&D, and two VV sectors has been repaired on schedule. The new tokamak construction procedure with the simultaneous welding of all 9 VV sectors has been decided. Preparation of the plant support systems is going well also. For example, the refrigerator has started operation. Using it, the Superconducting coil cold test (some TFs and a PF) will start from fall 2025. At the Neutral Beam Test Facility (NBTF), operation of SPIDER and commissioning of MITICA are underway.
In 2024, the ITER Council endorsed the overall approach of the ITER New Baseline with the main target dates of cryostat closure in 2033, start of DD H-mode operation in 2035, full plasma current and toroidal field operation in 2036, and start of the DT phase in 2039. This New Baseline is a comprehensive and feasible plan for assembly, integrated commissioning and operation developed so to keep the already agreed final project goals and to deliver the key objectives of ITER as early as possible. It takes a stepwise safety demonstration and licensing approach. The first operation phase is a scientifically meaningful research phase with sufficient heating power and divertor for starting DD operation and demonstrating the integrated fusion system with the nominal magnetic energy. The New Baseline is supported by the most updated scientific knowledge. One of the consequences is the change of the first wall armor material from Be to W. In order to maximize the efficiency of ITER project, further collaboration with the world devices is indispensable.
Towards DEMO, it is extremely important to 'evaluate' ITER through component manufacturing, tokamak assembly, operation, and experiments. Then ITER should contribute to establishment of fusion codes & standards and fusion regulation. ITER will serve as a platform for the development of next generation fusion human resources. Collaboration with private fusion companies is also an important role of ITER.

Integration of Waste Management and Decommissioning in European DEMO design

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Wednesday Parallel 1a - Safety, Regulation, and Neutronics I, Kresge Main Theater (Building W16, upstairs), June 25, 2025, 10:30 AM - 12:00 PM

The European DEMO aims to demonstrate the production of net electricity, the feasibility of operation with a closed-tritium fuel cycle and the adoption of maintenance systems capable of achieving adequate plant availability while demonstrating the safety and environmental advantages of fusion power compared with fission designs. Therefore, a coherent strategy for dealing with the wastes arising from the reactor needs to be part of the justification for realizing DEMO. This strategy needs to include the assessment of the wastes that will be produced during operation as well as decommissioning and their management from the production area to its final destination (clearance, recycling or final disposal), taking into account on-site (hot cells, radioactive waste facilities) and off-site facilities.
Although the location of EU DEMO has not yet been selected, and the local regulations and waste acceptance criteria cannot yet be considered, these aspects are being incorporated into the design from the earliest stages. The aim of this paper is then to present how waste management and decommissioning are implemented through guidelines and requirements to support designers. The following topics are being considered: selection of materials (including impurities), layout design and accessibility of components, standardization of equipment and requirements (involved both for maintenance and decommissioning), decontamination and waste management, compatibility with dismantling and decommissioning techniques …
Based on the guidelines from the waste management and decommissioning exercises, pre-conceptual design studies of the hot cell facility have started. The strategy that has been adopted relies on an effective maintenance scenario and the modularity of the structures (components and on site treatment facilities) to cope with the evolution of the needs of the hot cell facility from the early stages of the operation to the decommissioning. The first outcome of these studies shows that the modularity of the in-vessel components themselves (in particular the breeding blankets) needs to be further analysed to simplify the design and operation of the hot cell facility and thus reduce its cost.

Identifying Appropriate Regulatory Frameworks and Licensing Processes for Commercialization and Deployment of Fusion Power Plants

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Wednesday Parallel 1a - Safety, Regulation, and Neutronics I, Kresge Main Theater (Building W16, upstairs), June 25, 2025, 10:30 AM - 12:00 PM

Successful deployment of commercial fusion technology requires, in part, the creation of regulatory frameworks that enable effective, efficient, and predictable licensing. These frameworks must ensure public health and safety, protect workers and the environment, and enable the timely, repeatable, and economically competitive construction, operation, and decommissioning of commercial fusion power plants. Previous discussions on the regulation and licensing of commercial fusion technology have focused on the adaptation and use of existing regulatory frameworks for fusion technology. Debates have largely focused on adapting either the existing fission power plant regulatory frameworks or the existing non-fission radioactive material regulatory frameworks to effectively license fusion technology. This discussion typically focuses on the technology characteristics (e.g., can a fusion machine be defined as a particle accelerator) as the basis for selection of a regulatory framework without focusing on the underlying characteristics of the regulatory framework that determine the effectiveness of the licensing process. This work discusses the underlying characteristics of different regulatory frameworks (including those used for non-nuclear energy production, chemical production, and aerospace) and how they develop and ensure the safety case for different technologies. The regulatory frameowkr characterization is then applied to demonstrate how preferred regulatory frameworks could be identified based on proposed commercial fusion technologies, the level of design and operational information available, and the expected development and deployment timeline for the industry. Most robust characterization of different regulatory frameworks can help commercial fusion companies, regulators, and policymakers select or develop regulatory frameworks or create an evolving system of regulations that will enable the most effective, efficient, and predictable licensing and deployment of commercial fusion technology.

Fusion Regulation in the United Kingdom

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Wednesday Parallel 1a - Safety, Regulation, and Neutronics I, Kresge Main Theater (Building W16, upstairs), June 25, 2025, 10:30 AM - 12:00 PM

The development of fusion energy presents particular challenges, requiring regulatory frameworks that are both proportionate and flexible. These frameworks must ensure health, safety, environmental, and security protections, whilst maintaining public confidence and enabling technological advancement.

Collaborative dialogue among regulators, policymakers, industry representatives, researchers and others is integral to regulation across all hazards in the United Kingdom (UK). Central too is the right balance between a largely outcome-focussed and proportionate regulatory framework that facilitates, rather than stifles, innovation whilst maintaining good safety and environmental standards.

The application of this approach to the rapidly developing global fusion sector, which has positioned the UK as one of the leaders in the regulation of fusion, is the focus of this presentation from the UK regulators.

The Relationship Among Stoichiometry, Microstructural Evolution and Tritium Release in Ternary Lithium Ceramics

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Wednesday Parallel 1b - Blankets and Tritium Breeding II, Sala de Puerto Rico (Building W20 Room 202), June 25, 2025, 10:30 AM - 12:00 PM

One key feature of any D-T fusion breeder blanket is rapid and complete release of tritium to sustain energy production. In solid breeders, previous studies and the present work have noted a correlation between stoichiometry and tritium retention/release in Li ternary oxides and Li2O. Atomistic modeling provides insight into the fundamental mechanisms responsible for this behavior. In the radiation-damaged lattice, tritium prefers to reside on Li vacancies, diffusing through the material by hopping to nearest-neighbor Li vacancy sites. At the same time, the strongest trapping mechanism for tritium in these materials is O-T bonding. Li-rich ternary oxides tend to have few oxygen sites near the Li sites, while the opposite situation exists in Li-poor ternary oxides. Thus, it is much easier for tritium to diffuse through Li-rich oxides than Li-poor oxides.

For the past ten years, a comprehensive experimental and computational research program to improve fundamental scientific understanding of LiAlO2 and LiAl5O8 irradiation performance for tritium production in light water reactors has been pursued. For this application, the goal is to retain as much tritium as possible in the breeder ceramic. A similar effort has been underway for about five years to study irradiation performance of Li4SiO4, Li2SiO3, Li5AlO4, Li8ZrO6 and other octalithium ternary oxides for fusion solid breeder applications, where the desire is to release as much of the tritium as quickly as possible. The presentation will summarize results that are providing insight into the relationship among stoichiometry, microstructural evolution and tritium retention/release for all these compounds. Methods for building this understanding include both ion and fission neutron irradiation, followed by post-irradiation microstructural characterization using electron microscopy, x-ray diffraction, atom probe tomography, and secondary ion mass spectrometry. Atomistic modeling using density functional theory and molecular dynamics provides insight to help interpret the experimental results.

The work has shown that Li-rich materials effectively release tritium to a much greater degree than Li-poor materials, but with a tendency toward greater microstructural disorder that could jeopardize structural integrity. Specifically, these materials have demonstrated amorphization and Li loss in ion irradiation studies, as well as degradation caused by exposure to ambient moisture and CO2. Some of these effects could possibly be mitigated by blanket design, and some may be due to non-prototypic high damage rates during ion irradiation, but tritium-permeable coatings may help minimize deleterious effects caused by atmospheric exposure, while not hindering tritium release for fusion energy applications. The presentation will summarize conclusions to date and future studies needed to address existing knowledge gaps and identify an optimum balance between tritium release and microstructural stability for fusion energy applications.

Toward an environmental life cycle assessment of a DEMO-class tokamak fusion reactor

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Wednesday Parallel 1b - Blankets and Tritium Breeding II, Sala de Puerto Rico (Building W20 Room 202), June 25, 2025, 10:30 AM - 12:00 PM

The development of large-scale commercial fusion energy is often seen as a critical enabler of sustainable development over the coming decades due to its many potential environmental, economic and societal benefits. However, while current and near-term energy generation technologies are frequently assessed by techniques such as environmental life cycle assessment (LCA), the literature reveals that this has not yet been the case for fusion energy, predominantly due to its stage of development at scale, complexity, and variety of potentially deployable designs.
This work reviews the available literature in this area, identifying previous carbon footprint estimates of ~24 g CO2 kWh-1 which are not aligned with current standards in LCA, lack important data inputs, and do not consider the broad range of environmental metrics available via current impact assessment methodologies. Further, it identifies the steps needed to enable a usefully robust LCA of tokamak-based reactor designs that could inform technological development and policy by considering a broad range of likely environmental impacts over the reactor life cycle, from cradle to grave. The benefits of such an approach would include potential optimisation of resource usage, benchmarking of alternative designs, and communication of sustainability credentials.
The work also introduces current developments being undertaken to produce such LCA models, focusing on breeder blankets for DEMO-class reactors with consideration of the relevant materials, masses and estimated thermal and electrical energy outputs.
Lithium enrichment for tritium breeding is identified as a key area for further study from an environmental perspective, as current methods are either historic (such as the COLEX process) or actively under development at laboratory scale, and may require energy and material inputs with major implications for the resulting environmental impacts of the plant over its life cycle. Current ongoing work investigating this area from an environmental sustainability perspective is presented.
Our study offers a pivotal first step toward the incorporation of sustainability assessment into the development of fusion energy, providing actionable insights for technical research, policy, and the benefits of interdisciplinary approaches.

Project VICE (Validation of Ceramic Experiment): qualification of lithium ceramic breeders

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Wednesday Parallel 1b - Blankets and Tritium Breeding II, Sala de Puerto Rico (Building W20 Room 202), June 25, 2025, 10:30 AM - 12:00 PM

A key unknown for lithium ceramic-based breeder blankets is whether a practical blanket configuration can be made which will achieve a tritium breeding ratio (TBR) sufficient to support operation of a fusion power plant. Initial analysis by Oxford Sigma (internally and from literature (Giancarli et al., 2024)), suggests that it is theoretically possible but there remain a large number of practical uncertainties which bring doubt to achieving the modelled TBRs. The TBR calculations are normally based on neutronic modelling with Monte-carlo codes such as OpenMC. Project VICE (Validation of Ceramic Experiments) by Oxford Sigma, sponsored by the United Kingdom Atomic Energy Authority (UKAEA) Lithium Breeding Tritium Innovation (LIBRTI) Programme, is an experiment that quantifies these uncertainties to address the key unknown and increases confidence in predicting TBR calculations in any fusion tritium-deuterium fuel cycle-based device. Project VICE has the following objectives: 1) demonstration of lithium ceramic manufacture, 2) demonstration of tritium production, 3) demonstration of lithium enriched ceramics, 4) demonstration of tritium recovery, and 5) validation of modelling. Achieving these objectives will aid substantiation of the breeding performance and validation of tritium recovery modelling. The production of sufficient, high quality, and standardised data is required to qualify claims of material breeding performance. The primary objective of VICE is to demonstrate a scalable supply of lithium ceramic pebbles using existing capabilities, building upon existing breeder experiments in manufacturing at Oxford Sigma and mature existing programs on qualification of breeding performance. The program started in January 2025 and will conclude in March 2026 and this talk will provide an update on the program, scientific approach, detailed modelling, experimental plan, and expected data.

Progress and Outlook for De-risking FLiBe Salt Systems for ARC

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Wednesday Parallel 1b - Blankets and Tritium Breeding II, Sala de Puerto Rico (Building W20 Room 202), June 25, 2025, 10:30 AM - 12:00 PM

The ARC fusion power plant will produce 400 MW net electric and is targeting operation in the early 2030s at a site in Chesterfield County, Virginia. ARC will deploy a first-of-a-kind FLiBe salt blanket for the purposes of tritium breeding, component cooling and shielding, and heat removal. Carrying out these functions requires both specially adapted equipment for salt pumping, heat exchange, and tritium extraction, but also an overall salt chemistry strategy that manages corrosion, tritium speciation, and activation together. To enable ARC deployment in the early 2030s, CFS has begun a program of salt R&D work. This program involves understanding fundamental salt chemistry and behavior as well as maturation of novel key salt components. This work will share progress to date, provide an outlook on the upcoming roadmap for salt development, and highlight opportunities for synergistic public-private partnerships.

The engineering evolution of the compact high-field spherical tokamak ST40

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Wednesday Parallel 1c - Tokamak and Non-Tokamak Fusion Experiments, Kresge Little Theater (Building W16, downstairs), June 25, 2025, 10:30 AM - 12:00 PM

ST40 is a spherical tokamak, of 40cm major radius, 2.1 T toroidal field, 0.8 MA plasma current, with 1.8 MW of neutral beam heating and a plasma flat top duration around 150 ms. A 1 MW, dual-frequency (104/137 GHz) gyrotron is currently being installed for future provision of electron heating and current drive.

ST40 has a short history by the standards of most operational tokamaks, having been conceived, designed, built, operated, rebuilt and operated again entirely within the last ten years. It has also been near-continuously upgraded in this time. Future plans include a multi-year programme exploring the effects of lithium coated plasma facing components.

The purpose of ST40 is to expand the ST physics basis to higher magnetic fields and demonstrate the viability of compact, high-field devices. The choices made to meet this brief will be covered in this contribution.
A high level description of the machine architecture will be given with particular attention paid to the design of the toroidal field coil and its associated structure. The rapid initial builds of the machine gave an unrealistic impression of how easy major maintenance would be; modifications made in support of maintainability will be presented.

Many of the major upgrades carried out over the years were not originally envisaged. Some of these upgrades and the advantages and disadvantages of integrating them into an existing machine will be considered.
Operational experience with ST40 has shown the value of condition monitoring, combined with predictive modelling, for informing operational limits and repair plans. Some examples of cases where additional types of monitoring could have been useful will be given.

Finally, plans for ST40’s continuing evolution will be presented. Improvements foreseen in the near future include: structural modifications to increase reliability; addition of core fuelling via a pellet injector; replacement of the existing graphite limiters to facilitate lithium evaporation experiments; addition of radiofrequency heating and current drive; and enhancements in plasma diagnostics, along with a number of other improvements.

Real-time implementation of ITER first wall heat load control modules on TCV

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Wednesday Parallel 1c - Tokamak and Non-Tokamak Fusion Experiments, Kresge Little Theater (Building W16, downstairs), June 25, 2025, 10:30 AM - 12:00 PM

In this work, we present the implementation on the TCV tokamak of real-time modules for First Wall Heat Load Control (FWHLC) that have been developed for ITER, leveraging the shared Plasma Control System Simulation Platform (PCSSP)/Simulink-based Control Design and Deployment Suite (SCDDS) framework. This implementation demonstrates the flexibility of such an architecture for porting advanced control functions across different tokamaks.

As part of the defense-in-depth approach for ITER FWHLC, a real-time (RT) estimator for the heat flux impacting the plasma-facing components (PFCs) will work in tandem with the extensive ITER infrared (IR) camera monitoring system for early identification of plasma-generated hot spots on the tokamak first wall (FW). The main scope of the heat flux estimator is mapping the power entering the scrape-off layer (SOL) onto the shaped FW in RT, anticipating the temperature increase monitored by the IR system. A key innovation of this PCSSP module is the ability to evaluate in RT the plasma wetted area, exploiting a specially designed ray casting algorithm which combines RT magnetic equilibrium reconstruction with the 3D PFC structure. We demonstrate that this lightweight control-oriented module, adapted to both ITER and TCV geometry, approximates remarkably well in both cases the results of SMITER, a higher-fidelity, computationally intensive magnetic field line tracing tool.

Our experimental validation begins by comparing the RT heat flux estimates with offline reconstructions from IR cameras in dedicated TCV plasma discharges. Excellent agreement is found, particularly for limiter plasmas, confirming the accuracy of the estimator despite the simplifying assumptions required for its RT implementation. This study highlights the importance for heat flux estimation of incorporating the geometry of shaped PFCs and accounting for the SOL power spreading onto components parallel and perpendicular to the magnetic field lines.

In further experiments, the real-time heat flux estimation, now integrated into the TCV digital control system, triggers a transition to safer plasma configurations during discharges that exceed a predefined FW heat flux threshold. The new configuration is realized by the novel TCV shape controller, effectively acting as an inner control loop for FWHLC. Several prototype control solutions are tested to move diverted plasmas away from the TCV central column when excessive heat flux is detected following predesigned plasma radial displacements. These initial tests pave the way for the development of more refined FWHLC functions.

The integration of ITER FWHLC modules into the TCV digital control system and the successful experimental validation of the heat flux estimator demonstrate the potential for the rapid adaptation of ITER control functions on other tokamak control systems sharing the PCSSP/SCDDS technology. This work represents the first application of such an approach, constituting an example for future cross-platform implementations and supporting the development of FWHLC for ITER.

Plasma Operational Simulation (POPSIM): a Framework for Data-Driven Simulation and Control

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Wednesday Parallel 1c - Tokamak and Non-Tokamak Fusion Experiments, Kresge Little Theater (Building W16, downstairs), June 25, 2025, 10:30 AM - 12:00 PM

POPSIM is a control-oriented simulation toolbox developed in the machine learning framework JAX [1] to enable new capabilities for plasma control and tokamak operations: rapid simulations performed between pulses in operations, adaptation of models to incoming data from the tokamak, and the optimization of trajectories and controllers. Recent machine learning frameworks such as JAX enable the development of modules that span the full spectrum from highly structured (e.g. physics models with free parameters) to highly unstructured (e.g. neural networks) models. These capabilities enable the development of tools that combine physical principles with neural networks, which recent work has shown to be effective for predicting plasma dynamics [2]. With this in mind, POPSIM adopts the Scientific Machine Learning (SciML) paradigm (an extension of classical system identification), which leverages physical principles but prioritizes the predictiveness of experimental data. Taking inspiration from the classical control framework Simulink, POPSIM provides tools for system modelling via simulation and control modules. We report on progress training POPSIM modules to simulate and control aspects of plasma dynamics, including magnetic equilibria and transport, at Tokamak à Configuration Variable (TCV), and the development of POPSIM modules for SPARC real-time and off-normal events simulation.

[1] Bradbury, James, et al. "JAX: composable transformations of Python+ NumPy programs." (2018).
[2] Wang, Allen M., Darren T. Garnier, and Cristina Rea. "Hybridizing physics and neural odes for predicting plasma inductance dynamics in tokamak fusion reactors." NeurIPS Workshop on Tackling Climate Change with Machine Learning (2023).

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JT-60SA comes of age

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Wednesday Parallel 1c - Tokamak and Non-Tokamak Fusion Experiments, Kresge Little Theater (Building W16, downstairs), June 25, 2025, 10:30 AM - 12:00 PM

The JT-60SA superconducting tokamak successfully started operation in 2023, producing controlled mega-ampere diverted plasmas. This operation confirmed the successful integration of the numerous systems contributed since 2007 by Europe and Japan under the Broader Approach Agreement.

The record volume of 160 m^3 makes the tokamak plasmas produced by JT-60SA (R=3m, a=1.2m) the largest yet and the commissioning of JT-60SA addresses many challenges faced by the new generation of large devices – such as insulation integrity, plasma initiation, plasma control and disruption management.

Now in 2024-2025 substantial enhancements are being made to in-vessel components, heating systems and plasma diagnostics in preparation for high-power plasma experiments by the joint EU-JA Experiment Team.

23.5 MW of neutral beam heating are being installed, of which 10 MW will be provided by a 500 kV negative ion-based line and the rest by 4 tangential and 4 perpendicular positive ion-based units. The ECRH power available will be doubled to 3 MW from new steerable launchers.

Error field correction coils have been mounted inside the vacuum vessel and now the in-vessel winding of the fast plasma position control coils is underway. Stabilizing plates and resistive wall mode correction coils will be installed as well as a pumped carbon lower divertor and carbon first wall. Moreover, 15 additional diagnostic systems will be installed before operation restarts in 2026.

Further enhancements are planned for subsequent campaigns including more heating power, more diagnostics and tungsten plasma-facing components.

This overview will summarise both the JT-60SA story so far and plans for the future, when JT-60SA will provide unique results in support of ITER and DEMO operation.

Non-destructive examinations in support of the design qualification activities of the outboard first wall of DTT

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Wednesday Posters 1, Lobdell (Building W20 Room 208), June 25, 2025, 10:30 AM - 12:00 PM

The Divertor Tokamak Test facility (DTT) aims at investigating power exhaust solutions that can be relevant for DEMO and future fusion reactors. Moreover, it will contribute to DEMO by making the possible DTT scenarios relevant for future reactors, in terms of sufficiently long flat-top times at high currents and a full-tungsten (W) wall. Therefore, the first wall of the DTT is necessarily actively cooled and fully covered by W as armour material. Depending on their position inside the vacuum vessel, the FW modules can be distinguished into 3 sub-components, named Inboard First Wall (IFW), consisting of limiters and standard modules, Top First Wall (TFW) and Outboard First Wall (OFW) with different expected loads and, consequently, different design solutions. Each sub-component is hydraulically in parallel with respect to the others. The OFW design features actively cooled steel plates with integrated cooling circuits having a square cross section. Each module is fed in parallel with respect to the others, and cooling water is provided at 4MPa, 60°C and adequate mass flow rate for ensuring the nominal water velocity in the square conduit. In the context of the necessary qualification activities of the envisaged OFW design, samples and mock-ups have been fabricated using additive manufacturing techniques. Atmospheric Plasma Spray (APS) with local inert gas protection has been selected to deposit W coatings on the plasma-facing side and with a thickness up to few hundred microns. Following a preliminary process optimization and characterization of the coatings, aimed at minimizing porosity and oxide content while maximizing the bonding strength, medium-scale OFW mock-ups have been manufactured and W-coated, in order to be tested at thermal fatigue at the electron beam facility HELOKA by cyclic loading up to 5000 cycles at 0.5MW/m2.
The present work reports on the activities carried out at the Special Technologies Laboratory (TES) of ENEA Frascati in order to qualify the Ultrasonic Technique (UT) as a non-destructive testing method for the acceptance of these components during the production phase. The use of the UT to inspect thicknesses of few hundreds of microns is not trivial and requires a dedicated R&D phase and the development of a specific system and procedure. In detail, the UT equipment used in this work is based on the pulse-echo water gap method that was employed to assess the structural integrity of small samples and mock-ups after fabrication and after the Medium Heat Flux experimental campaign. Finally, metallographic inspection is also employed for a preliminary assessment and, at a later stage, for a final validation of the results obtained by UT.

Closed Loop with FPGA-based Full Digital Phase Comparators of Multi-chord Interferometer for Real-time Density Control in KSTAR

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A full digital-phase-comparator (DPC) has been implemented in the KSTAR interferometers to facilitate a real-time feedback control of multi-line densities : 5 tangential probing lines that passes through different tangential radii of the plasmas and an additional reference line that bypasses plasmas. Thus in total 6 lines of optical cords are to be measured robustly in real-time. In addition, each line has two mixed signals with different colors of lasers: 10.6 µm of infrared from a CO_2 laser and 660 nm of visible beam from a diode-pumped-solid-state (DPSS) laser respectively. This combination is essential to cancel out the vibration noise inherent in the two-color interferometry technique. Counting the entire optical lines, total 12 channels of digitizers are necessary. Each signal of 40 MHz of intermediate frequency (IF) is digitized by a Vadatech MRT-522 module. Digitized signals are directly routed to another Vadatech module AMC-522, which is equipped with a Xilinx FPGA together with 16-bit 160 MHz of DAC. The FPGA generates the phase signal in every 25 ns by implementing the CORDIC algorithm to calculate arctangent of the 40 MHz IF signals. To reduce a noise with stable operation, the raw signal is averaged down to 625 kS/s of final phase signals that provides video-bandwidths up to 313.5 kHz to sufficiently capture most of instabilities in the KSTAR plasmas. The phase signals are further calculated in its Host PC to provide the actual line-integrated (or averaged) electron densities in real-time. The calculated density is transferred to the plasma control system (PCS) directly via reflective memory (RFM) technique. The requested signal sampling from the PCS is 1 kS/s which is realized by taking a mean value of every 625 kS. The PCS finally calculates drift-corrected electron density by subtracting reference phases from each density of different chords. The compensated drift is mostly caused by thermal instabilities of the laser. Entire 5 different cords of real-time density provides real-time capability for density pedestal or even profile control for the future research as well as present line-density controls in many topics of physics research.

Initial commissioning of the Novatron N1 device

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Wednesday Posters 1, Lobdell (Building W20 Room 208), June 25, 2025, 10:30 AM - 12:00 PM

The Novatron [1] is a novel, axisymmetric, magnetic mirror-cusp concept. The magnetic field configuration produces favourable curvature at the outer plasma surface that should be inherently stable against interchange instabilities that have limited previous magnetic mirror devices.

This work focuses on the initial commissioning phases to qualify the device for experimental campaigns. It aims to present an outline of the integration and commissioning process with particular focus on the challenges and lessons learned during the integration of a first of a kind device being designed, built, and tested by a newcomer in the fusion research space. The commissioning process focused on three main steps: local system commissioning, integrated system testing, and finally first plasma operation.

The vacuum chamber is the structural backbone of the machine whose geometrical properties are dictated by the Novatron magnetic field configuration [1]. Three main sections, each composed of multiple segments, form a 2.7 m3 volume. All the vacuum segments, including the auxiliary in-vessel equipment, where tested and commissioned following a methodical approach that included cleaning, helium spectroscopy leak testing, residual gas analysis, and leak rate characterisation. The implementation of temporary covers to construct modular sub-chambers enabled the process to be executed incrementally. This process was fundamental in finding different leakage points in the chamber that could be addressed. The complete chamber could then be successfully assembled and pumped down to the UHV range.

Given that the magnetic field configuration constitutes the defining characteristic of this novel device, it was important to measure and characterize the correct correspondence between simulations and modelling with the physical device. The copper magnet coils were axially aligned to the vacuum chamber by iteratively measuring the field and mechanically adjusting their positions. The field was then characterised by probing with an instrumented probe via actuated vacuum bellows. The power supply system includes protection for withstanding sudden loss of power and the effect of mutual inductance between coils and was both bench tested and as a complete system.

The hydrogen gas for fuelling is puffed into the chamber via in-vacuum distributor rings that are meant to assure axisymmetric injection. Fast solenoid and pneumatic valves along with pressure and flow control systems were characterised and compared against simulations.

The first plasma ionisation and heating system uses ECRH implemented by 6 x 6 kW, 2.45GHz RF amplifiers evenly spaced around the chamber. Incremental and continuous testing and integration included building a separate small mirror device to test on magnetised plasma. This derisking and agile strategy has greatly contributed to the success of the N1 commissioning.

Integrated system testing was performed using an in-house developed, LabView based, experiment control system which coordinates input parameters and triggering for both machine systems and diagnostics. Following the validation of individual system requirements, the initial plasma tests were positive and laid the groundwork for the start of the experimental campaign.

[1] Jäderberg, J. et al., "Introducing the Novatron, a novel fusion reactor concept". Preprint, arXiv 2310.16711, 2023

2D Hall Arrays for High Resolution Tokamak Magnetic Field Imaging

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The use of a high-density two-dimensional array of Hall effect sensors is proposed to allow precise monitoring of the magnetic field in the upcoming AtomCraft student-built tokamak at UNSW, Sydney. The device is not intended to be plasma facing or to be used directly for plasma diagnostics, but instead to directly compare the observed magnetic field cross section with Multiphysics simulations. A first prototype has been built using entirely off-the-shelf components, using 256 sensors mounted on a 6-layer circuit board with surface mount components on both sides to achieve the compactness required to fit inside the Tokamak vacuum vessel. This prototype is capable of sampling at 40,000 samples-per-second, an order of magnitude faster than similar devices used outside of fusion. This massive increase in sampling rate is achieved through parallelising both the analogue and digital components of the system, using simultaneously-sampling ADCs to reduce the requirement for series multiplexers and FPGAs respectively. The fusion-focused design also incorporates increases resistance to electromagnetic interference, employing shielding for sensitive components and active analogue filters that also work to prevent aliasing. The device can measure magnetic fields up to ±150 mT, with an RMS noise floor below 1 mT and an analogue bandwidth of 20 kHz, which is suitable for the AtomCraft tokamak with an on-axis toroidal field of 100 mT and a pulse length of 100 ms. The device can also run in a 'real-time' capability as a magnetovision camera, sacrificing sampling rate for a lower noise floor through hardware averaging and oversampling. Early testing has shown good resistance to EMI and generally favourable device behaviour, with full testing tied to the ongoing development of the AtomCraft Tokamak.

If practical, a live demonstration of the device could accompany a poster.

Minimizing the Electromechanical Stresses in Poloidal Field Coils by Optimizing their Number, Location, and Structure using FREDA Framework

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Poloidal field (PF) coils play a crucial role in sustaining equilibrium and preserving the shape of highly confined and magnetohydrodynamic (MHD) stable plasmas. Minimizing the electromechanical stresses in the PF coils depends on minimizing the electric currents and hoop and center forces on these coils. This requires an optimization of the PF coils number, sizes, structures, and locations within the vacuum chamber to maintain the same equilibrium at lower currents. A free-boundary MHD equilibrium code – FreeGS, is employed to construct the plasma equilibrium based on the configuration and currents in PF coils. On the other hand, an integrated modeling framework, such as the IPS-FASTRAN, is required to estimate the transport coefficients, confinement factor, and MHD stability in a plasma configuration. Combining these capabilities under the Fusion REactor Design and Assesment (FREDA) whole facility modeling and optimization via advanced simulation of integrated physics and engineering modeling framework provides a unique capability to design high performance plasma scenarios consistent with the engineering constraints in tokamaks. Here, we present the capability of the FREDA framework which includes the FreeGS workflow that was utilized to minimize the currents, forces, and electromagnetic stresses on the PF coils by optimizing their number, structures, and locations while maintaining MHD stable plasma configuration of large confinement factor. The workflow is initialized with a configuration of plasma parameters and coils' locations. FreeGS finds the initial equilibrium at the minimum total current in PF coils. Hence, FreeGS optimizer minimizes the currents and hoop and central forces on the PF coils while maintaining the initial equilibrium. Finally, the input configuration is updated with the optimized configuration. An iterative process will be implemented to find the optimum configurations for all plasma equilibria over the ramping up phase of burning plasma operation. The optimum coil configuration for predefined plasmas will be stored in a database for designing plasma shape controllers using machine learning and artificial intelligence tools in future work.

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The Superconducting Magnet Test Facility at the MIT Plasma Science and Fusion Center

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The Superconducting Magnet Test Facility (SMTF) at MIT's Plasma Science and Fusion Center provides a range of capabilities to support development of HTS magnet technology. Constructed between 2019-2021 around the SPARC Toroidal Field Model Coil (TFMC) – a non-insulated HTS magnet – it has subsequently been upgraded and employed to service a broader range of test objectives, including those of insulated magnet systems. The main facility provides a large cryostat with power supply and binary HTS current leads capable of continuous operation at 50 kA. Cooling is provided by a closed-loop supercritical helium cryogenic infrastructure that provides ~600 W of cooling at 20 K. The facility also provides a large amount of instrumentation for data acquisition and control. The SMTF has been used to test the SPARC toroidal and central solenoid model coils, as well as a smaller forerunner to the CSMC, and will host the "Magnet 0" test coil of Type One Energy in 2025. In support of these later experiments, a quench detection and fast discharge system has been deployed, with response times on the order of 100 ms. Presently, discharges are limited to 125 V, with further hardening of the system desired to increase this ceiling. Moreover, proposals have been prepared to install a set of pulsed power supplies at 800 V and 16 kA, together with new current leads and other ancillary systems.

A satellite facility services testing in liquid nitrogen. A 16 kA, 10 V power supply is available for testing a wide variety of coils, cables, and other samples. Several liquid nitrogen cryostats host these samples, and custom cryostats may be built for form factors that exceed the parameters handled by these, while direct connection to a 26000-liter (7000 gallon) LN2 tank provides for substantial volumes of liquid nitrogen for long-duration, large-scale tests. Desired upgrades to this facility include an expansion of the instrumentation and control system, the addition of a small cryostat and helium circulation loop for test articles of a scale not appropriate to the larger facility, and a further increase in available voltage and fast discharge capabilities.

Conversational Agents based on Large Language Models for Tokamak operations Data Retrieval and Analysis

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Wednesday Posters 1, Lobdell (Building W20 Room 208), June 25, 2025, 10:30 AM - 12:00 PM

The complexity of tokamak operations generates extensive datasets and knowledge, presenting challenges in data retrieval and decision making processes, both before and during tokamak operation preparations. Existing information retrieval solutions require dedicated in-situ tools that are based on the direct interaction with a database. These tools are not always optimised for global data analysis, can be a steep learning curve for researchers, and may not be fully suitable for Scientific Leaders (SL) during the dynamic environment of tokamak operations. To address these challenges, building specialized AI tools can accelerate data access and information search, thereby optimizing tokamak operations. Recent advancements in artificial intelligence (AI), particularly Large Language Models (LLMs) and conversational AI, offer a promising framework for more efficient data management. This work proposes AI conversational agents specifically designed for tokamak operations. leveraging large language models (LLMs) with domain-specific knowledge in plasma physics and fusion, these agents facilitate data retrieval through intuitive querying, contextual understanding, efficient data analysis, and data updates. We present a set of case studies highlighting the utility of developing AI assisted systems in order to accelerate research workflows and supporting tokamak operations. This work is currently under development as part of the EuroFusion work package Preparation of ITER Operation (WPPrIO). The proposed AI system interprets natural language queries from operators, allowing faster access to experiment information within a conversational format. It retains conversation history for efficient data exploration and searches localized datasets containing tokamak operational data, similar experimental results, diagnostic readings, and operational contextual events such as disruption. The system then provides responses supplemented by visualizations and summaries of key information, to get expert input and enhance the system answers. We also explore the challenges faced in designing the system, such as ”hallucination”, and discuss how integrating domain-specific knowledge and robust reasoning capabilities ensure accurate data analysis. The resulting conversational AI system is expected to enhance data-driven decision-making and improve operational performance of tokamak systems.

Design and Verification of Water Leakage Suppression Scheme for EAST

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Wednesday Posters 1, Lobdell (Building W20 Room 208), June 25, 2025, 10:30 AM - 12:00 PM

During the long-pulse high-parameter experiments of Experimental Advanced Superconducting Tokamak (EAST), several in-vessel loss of coolant accidents (LOCAs) occurred, forcing the experiments to be halted. Once the coolant intruded into the vacuum, it immediately flashed, causing the vacuum pressure to rise rapidly, posing a threat to the vacuum pumping system and the safety of the device. This paper presents a design scheme for water leakage suppression in EAST. Numerical analysis has been conducted using RELAP5 to discuss the feasibility of the scheme, and further experimental research has been carried out. The leakage suppression scheme involves triggering a stop signal for the cooling water system (CWS) based on vacuum pressure signals, and installing pneumatic shut-off valves and check valves at the upstream and downstream of the plasma facing components (PFCs) to isolate the coolant, thereby achieving the function of leakage suppression. The results of the numerical analysis indicate that the mass flow rate of the leakage water decreases rapidly after the pump is stopped, and the closure of the shut-off valve ensures that the leakage flow rate is maintained at an extremely low level. The pressurization rate and the amount of leakage water are effectively suppressed. The experimental data show good consistency with the numerical results, demonstrating the feasibility of the leakage suppression scheme design and equipment selection. This study provides a reference for the upgrade of the EAST CWS, as well as for the CWS design of future fusion reactors.

Divertor Monoblock Multiphysics Analysis

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Advanced computational tools play a crucial role in ensuring the rapid deployment of fusion energy systems due to the multiphysics interactions occurring at the component level. For example, plasma-facing components (PFCs), such as the divertor, undergo thermal loads and stresses, nuclear heating from neutrons and ions, and conjugate heat transfer in the solid material regions and water-cooling channels. High-fidelity multiphysics computational tools allow scientists and engineers to better understand the complex physical phenomena occurring in these components, ensuring their appropriate operation within fusion energy systems. To address these challenges, the Multiphysics Object Oriented Simulation Environment (MOOSE) framework is being leveraged for fusion modeling and simulation through the MOOSE-based Software for Advanced Large-scale Analysis of MAgnetic confinement for Numerical Design, Engineering & Research (SALAMANDER) code. SALAMANDER is designed as a multiphysics and multiscale computational tool capable of 3D high-fidelity fusion system modeling, and it leverages modular MOOSE-based physics capabilities such as thermal mechanics, heat transfer, and fluid dynamics. In addition to MOOSE, the MOOSE-based application Cardinal is employed by SALAMANDER for neutronics capabilities and the MOOSE-based Tritium Migration Analysis Program, version 8 (TMAP8) is used for tritium transport and fuel cycle simulations. Finally, the MOOSE-Cardinal-TMAP8 coupling possesses the capabilities for plasma edge modeling to simulate plasma-material interactions for PFCs.

This work aims to leverage SALAMANDER’s capabilities to conduct a case study on an ITER-like divertor monoblock. The divertor is responsible for managing and removing excess heat and impurities from the plasma, protecting the tokamak's components, and maintaining plasma stability during operation. The developed model will utilize its multiphysics capabilities to simulate the neutron interaction from the plasma and the thermal-hydraulic effects from the cooling channel on the divertor monoblock. The neutronics analysis employs a quasi-static approach to calculate heating from a Monte Carlo neutron transport simulation, where a planar neutron source is defined at the top boundary of the divertor monoblock. The same pulsing function is then used to normalize the OpenMC tally results. The thermal-hydraulics analysis utilizes the Reynolds-Averaged Navier-Stokes (RANS) approach, implemented using the Navier-Stokes module in MOOSE, to model fluid flow through the divertor cooling channel. The heat transfer between the solid divertor monoblock and the water-cooling channel is handled using an interface kernel. Tritium is modeled using TMAP8 and accounts for diffusion, trapping, and solubility. By incorporating advanced modeling tools into a fully unified framework, SALAMANDER serves as a vital resource for advancing the deployment of fusion energy.

STRUCTURAL ASSESSMENT OF IN-VESSEL ELM CONTROL COILS FOR THE TCABR TOKAMAK

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The Tokamak à Chauffage Alfvén Brésilien (TCABR) will be upgraded to be capable of studying the impact of resonant magnetic perturbation (RMP) to control edge localized modes (ELM). For this, 108 independently powered in-vessel RMP coils are being designed. These coils are divided into two groups of 54 coils. The first group is located on the high-field side (HFS), and is known as the CP-coils. The second group is located on the low-field side (LFS) and is called the I-coils. Each group is organized into three rows, with each row containing 18 coils, enabling operation with a toroidal mode number, ntor ≤ 9. These coils were first designed through magnetohydrodynamic (MHD) simulations conducted using the non-linear two-fluid resistive MHD code M3D-C^1 considering the plasma response to the applied RMP fields. As a result of this first analysis, the electrical currents needed for ELM control and the coils' dimensions and positions within the vacuum vessel were determined. These coils will operate with both direct (up to 60 kA-turn) or alternating (up to 30 kA-turn) current, high voltages (up to 4 kV), and frequencies up to 10 kHz. They must withstand relatively high temperatures (up to 200 ◦C), high vacuum conditions (p ≤ 1 × 10−7 mbar), strong magnetic fields (2.25T), and support substantial electromagnetic force densities (0.2 N/mm3). Given that the coils will be installed inside the TCABR vacuum vessel, the materials and processes employed in their fabrication must be compatible with a high vacuum. To satisfy all these conditions, a Vacuum Pressure Impregnation (VPI) process was applied to fabricate all the coil packs. Besides, the structural case and the welding processes were also designed to accomplish the vacuum requirements. To evaluate the mechanical design several finite element simulations using the ANSYS modules, such as Magnetic, Static Structural, Thermal, Modal, and Harmonic Analyses, were carried out. The results of stresses were compared to the allowable stress limit of the material based on both ASME and ITER criteria, showing that at least one criterion was met. Furthermore, experimental tests are being conducted to compare the finite element results with experimental data. These tests will measure the natural frequencies, temperature increase, and structural strain in the structural case under the TCABR toroidal magnetic field.

Deep Learning for Intelligent Monitoring of the WEST Tokamak First Wall Using Infrared Imaging: An Overview

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First wall protection is a critical challenge in fusion machine operation, requiring fast and precise diagnostic capabilities. The infrared diagnostics of long-pulse fusion machines, such as the WEST tokamak, generate a large amount of video data. It is a challenge for human operators to conduct real-time and post-pulse thermal scene analysis.

The real-time protection has traditionally relied on thresholding techniques and conventional image analysis methods. While these approaches are both robust and effective, they have certain limitations. For instance, they lack adaptability to newly emerging hot spots and require pre-defined regions of interest (ROIs). Any displacements of the line of sight, such as those caused by camera or tokamak movement, can compromise the protection system, making it ineffective. Consequently, deep learning-based methods have rapidly emerged as promising alternatives, offering significant advantages in areas where conventional approaches fall short.

The aim of this work is to describe and review the complete framework (hardware and software) developed through the years to monitor and analyse the first wall of the WEST tokamak using infrared imaging data. The developed tools are categorized into two primary types: active real-time methods designed to interact with the plasma control system, and decision-support tools intended for post-pulse analysis.

The real-time analysis methods include two detection models (yes/no at image scale): one for electric arcs and another for runaway electrons. Both methods, similar in structure and implementation, are optimized to run on CPUs, enabling their use for real-time plasma control. Notably, the arc detection algorithm has been deployed and linked to the PCS to regulate the power injected by the lower hybrid (LH) antennas during the recent C10 and C11 experimental campaigns. The model achieved an accuracy of 98% on C10, having detected only one false positive (post-disruption, therefore not limiting the operation). Another key aspect of real-time detection is the Thermal Event Detection model, which operates on GPUs to detect, track, and characterize thermal events in real time. This model addresses the limitations of fixed ROIs by enabling the tracking of precise hot spots as needed. In the long term, this model has the potential to generate adaptive ROIs, for instance, by integrating detection with pixel-level component mapping.

Post-pulse decision-support tools are just as important. Among them, a unidentified flying object (UFO) detection tool serves as an advanced method for identifying and tracking more than 97% of tungsten deposit particles visible on IR cameras. These detections can help in finding correlations between UFOs and plasma disturbances, providing valuable insights into the experimental conditions. In parallel a multi modal Large Language Model (LLM) is under development and evaluation in the institute. This multi modal LLM integrates expert knowledge for automated pulse summary analysis. This tool has been successfully deployed and actively used between pulses by infrared experts during the C10 & C11 experimental campaigns (10/2024 - 04/2025). The evaluation of the model by experts indicates that over 62% of the generated responses were satisfactory, with 50% of them being characterized as expected from such an analysis.

Magnet System Design and Optimization for A Novel Plasmoid Magnetic Reconnection Thruster Prototype

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The Plasmoid Magnetic Reconnection Thruster (PMRT), inspired by solar flare acceleration, stands as a notable innovation in plasma physics research. Stemming from fast magnetic reconnection studies conducted on the National Spherical Torus Experiment (NSTX) [1], a PMRT prototype is under development at Princeton Plasma Physics Laboratory (PPPL), aiming to generate thrusts through continuously creating plasmoids, enabled by a specific configuration of a static magnetic field [2]. This static magnetic field is produced by a magnet system which serves as a core component of the plasmoid thruster and is essential to the occurrence of magnetic reconnection and the generation of plasmoids. This study focuses on the design and optimization of the water-cooled copper magnet system to produce a stable and strong magnetic field with desired spatial distribution required by the PMRT. The design process begins with the target magnetic flux distribution derived from magnetohydrodynamic (MHD) simulation. Linear programming is employed to identify the optimal magnet configurations and locations that minimize the input power, significantly reducing system costs. A parametric sweep further refines the magnet configuration and determines the geometric parameters of the magnets. The final magnet design is validated through MHD simulation, ensuring plasmoid generation and completing a close-loop engineering design process. In addition, the study addresses the integration of the magnets into the thruster prototype and confirms the structural integrity of the magnet system under operational conditions. Special attention is devoted to a conceptual design that incorporates superconducting magnets for future iterations of the PMRT to meet the stringent requirements of deep space exploration. This comprehensive approach delivers an economical and efficient design for the thruster's core component, establishing a foundation for the development of a new cost-effective and robust plasma propulsion technology.

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Data-driven multiscale fracture modeling of plasma-facing materials

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Plasma-facing materials (PFMs) in fusion reactors experience high heat fluxes, causing thermal fatigue and cracks that affect material lifespan. Despite advances in fracture modeling, challenges remain in characterizing microstructures, understanding their thermal shock response, and modeling crack initiation and growth. This presentation outlines our multi-scale fracture modeling approach for tungsten PFMs. Using machine learning and generative adversarial networks (GANs), we created a microstructure database based on two-dimensional (2D) scanning electron microscopy (SEM) data from the JUDITH facility. These 2D microstructures are converted to three-dimensional (3D) and incorporated into the Multi-Physics Lattice Discrete Particle Model (MP-LDPM) for modeling thermo-elastic and plastic deformation. The results are then integrated into the CabanaPD fracture code to simulate macroscopic failure. Ongoing development aims to link MP-LDPM and CabanaPD for predicting both thermo-plastic and brittle behavior, with experimental validation underway. This multi-scale framework seeks to improve the reliability and lifespan of PFMs under extreme heat flux conditions.

Thermo-Mechanical Analysis of W7-X Control Coils for High DC Pulse

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The control coil (CC) system of Wendelstein 7-X (W7-X) consists of ten identical 3D shaped coils, situated behind the baffle plates of each corresponding divertor unit, designated to both, rectify the error field and sweep hot spots on the divertor target plates. These coils are constructed by winding hollow copper conductor (15 mm x 15 mm square with Ø 8 mm water channel) into eight turns (double pancake). Turn, interlayer and ground electrical insulations form a winding pack (WP). Each WP is embedded in a stainless steel (SS) coil case, which is attached by three supports, one fixed and two sliding, to the plasma vessel (PV).

Enhanced plasma operation scenarios require implementing a 3.5kA direct current (DC) pulse to the CC system, which is 40% higher than the CC design value of 2.5kA DC. The original ANSYS CC finite element model (FEM) developed in 2017 [1] has been refined for both thermal and structural analyses to check the CC operation under higher thermal and electromagnetic loads. The analyses were performed to simulate CC pulses during two cases of plasma heating cycles accepted for 2024/2025 campaign: the short plasma pulse (the PV at RT) and long plasma pulse (of ca 200s, the PV average temperature is approx. 50°C).

The refinement included modelling the water flow through the hollow channel with the aim to capture thermal gradient between conductors and corresponding thermal stresses in the orthotropic insulation. To avoid artificial concentration of stresses due to coarse mesh, the structural model employs refined quadratic mesh on each component of the CC FEM. The mechanical analysis takes into account the deadweight, magnetic forces applied to the copper conductor and temperature gradient from the transient thermal analysis at the time point when the conductor temperature reaches its maximum.

As a result, short CC pulse with ~10s flat top of 3.5kA DC at 2.5T main magnetic stellarator field is compatible with the W7-X control coil power supply system. However, the analysis indicated some critical areas with the stress levels above allowable limits in the conductor, insulation and SS coil case. Therefore, two FE submodels with finer mesh have been created for the critical regions and analysed under boundary conditions interpolated from the original FEM. The submodelling results confirmed that the CC can be operated with the required short pulse when the PV temperature is below 50°C, without overloading of metallic components and delamination of the insulation.

Further CC analyses for next operation phases are also discussed in the paper.

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Liquid Metal In-Vacuo Injection (LIVIn) system to measure the permeability of porous structures for liquid metal plasma-facing components

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One of the major hurdles on the path to long-cycle fusion reactors is enhancing the durability of plasma-facing components to resist extreme conditions inside the reactor. In this context, liquid metal plasma-facing components (LMPFC) show remarkable advantages, with a renewable protective layer of liquid metals (LM) covering the underlying solid components. Also, they can mitigate steady-state heat flux through circulation and high heat fluxes in the case of transient events through evaporative cooling. Further, there is a potential to pump deuterium and tritium fuels if liquid lithium is used [1]. Among several configurations proposed for LMPFC, one involves having a capillary porous layer filled with LM facing the plasma while a stream of LM flows beneath, replenishing the porous layer. This configuration is beneficial in controlling the potential instabilities of the liquid layer as the surface tension forces between LM and the porous substrate can counteract forces leading to instabilities [1,2]. However, to assess the actual viability of this configuration, one must investigate the interactions between prospective liquid metals and the porous substrates.

Towards that end, the Liquid Metal In-Vacuo Injection (LIVIn) system is designed to study the mass transport of LM across porous substrates [3]. This system is fully compatible with UHV conditions and is capable of withstanding corrosive liquid metals such as Lithium. Also, the presence of a high-speed camera system allows it to measure the flow speed of the LM front. Other unique features include the in-vacuo LM seal, which allows efficient sample swapping, and a sealable metal loading capsule to avoid oxidation of the loaded metal. The LIVIn system is currently being tested and commissioned inside a test vacuum chamber. Tests conducted so far with liquid Lithium have confirmed the lithium pumping capability, proper functionality of the heating system, and the in-vacuo LM seal. Ongoing tests of the system focus on flow calibration to determine the system's limits, and the next phase of testing will be to establish the permeability measuring mechanism using porous samples. The latest test results and verified system capabilities will be presented.

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Dispersion Strengthened Tungsten: Response to H-Mode Plasma Exposure in DIII-D

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The surface stabilities of 1 wt% transition-metal carbide (TaC, TiC, ZrC) dispersion-strengthened tungsten (DSW) alloys have been evaluated under H-mode plasma conditions in DIII-D using the Divertor Materials Evaluation System (DiMES). For each DSW type, samples mounted flush with the divertor floor and samples mounted at a 10° angle towards the incoming plasma fluxes were exposed, and the outer strike point was rastered across the DiMES location to make the incident heat fluxes more uniform across the samples. The inter- and intra-ELM heat fluxes for the flush samples were 2MW/m² and 6MW/m², respectively, and for the angled samples were 15MW/m² and 42MW/m², respectively. Identical exposures and analyses were repeated for DSW which has been pre-implanted with 10^{18}/cm² helium, and while pre-heating the DiMES system to 500 °C. ITER grade tungsten samples were concurrently exposed as controls. Post-exposure analysis by scanning electron microscopy (SEM) found no evidence of melting or cracking on the flush-mounted samples, regardless of composition. However, there is significant cracking damage and evidence of dispersoid ejection on all angled DSW samples in each case: as-received, helium implanted, and heated. In addition to surface cracking and melting, the evolution of dispersoids is characterized via energy-dispersive spectroscopy, and x-ray photoelectron spectroscopy. A micron-scale fiducial marker was placed on each sample before plasma exposure, enabling direct comparison of the same dispersoid before and after exposure. An absence of melting or cracking on all flush samples suggest excellent surface stability of DSW. In the case of W-ZrC and W-TaC angled samples, damage was significant enough to destroy the fiducial marker in addition to micron-scale cracking. In contrast, the fiducial marker is visibly evident on the surface of all three W-TiC angled samples: as-received, He implanted, and heated. While leading edge effects led to macroscopic cracking on the top of the angled W-TiC sample, there is no evidence of melting.

Tungsten (W) is a leading candidate Plasma-Facing Component (PFC) for the divertor of tokamak reactors.[1] Transition-metal carbide dispersion-strengthened W alloys have shown elevated recrystallization temperatures, improved crack resistance, and higher strength compared to pure W.[2-4] W/carbide interfaces have been shown to act as effective defect sinks, suppressing helium induced nanostructuring.[5] This DIII-D experiment was the first time that carbide DSW alloys were exposed to divertor heat fluxes of this magnitude. These results are an important step to improve the technical readiness level of carbide DSW alloys, leading toward the down-selection of candidate PFC materials for Fusion Pilot Plants based on their heat flux response.

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CONJUGATE HEAT TRANSFER LARGE EDDY SIMULATION OF A HYPERVAPOTRON: FROM INCIPIENT NUCLEATE BOILING TO CRITICAL HEAT FLUX

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Efficient management of high heat fluxes in tokamak reactors poses a significant challenge critical to their sustained operation. A key solution lies in the use of specialized high-heat-flux devices like the HyperVapotron, integral to a tokamak's first wall, neutral beam injectors, and divertor systems. Although previous numerical investigations have predominantly relied on Reynolds-Averaged Navier-Stokes (RANS) models, these often fail to effectively capture the complex thermal-hydraulic phenomena associated with HyperVapotrons. It has been previously established in the literature that variations among RANS turbulence models can lead to significantly different flow patterns, underscoring the method's limitations.

In contrast, Large Eddy Simulation (LES) presents a more sophisticated turbulence modelling approach that could enhance our understanding of flow dynamics within HyperVapotrons. However, prior LES work has mainly focused on convective heat transfer, leaving a critical gap in the analysis of nucleate boiling and critical heat flux (CHF). Additionally, existing models for phase change processes in HyperVapotrons struggle to transition between various heat transfer regimes, limiting their applicability in automated design optimisation workflows. This highlights the need for advanced modelling techniques that can address the complex behaviours inherent to these devices.

This research aims to develop a comprehensive conjugate heat transfer simulation capability using LES, enabling seamless simulations across all heat transfer regimes without manual intervention. We will enhance the NekRS framework to incorporate models for multiphase flow and film boiling, allowing for high-fidelity simulations that accurately represent the full thermal-hydraulic complexities of HyperVapotron operation. A rigorous verification and validation process will be implemented to ensure the accuracy and reliability of the developed solver, with a particular focus on conditions typical of HyperVapotrons.

This work is expected to significantly contribute to the fusion community in several ways: (1) enhancing our understanding of heat transfer and turbulence dynamics related to HyperVapotrons, (2) providing a comparative analysis of LES and RANS methodologies to elucidate RANS model limitations, and (3) supplying valuable data to inform the development of improved RANS models. We believe these contributions will facilitate advancements in the design and optimisation processes of high-heat-flux devices utilised in fusion reactors.

Impact of high-power Cosh-Gaussian laser beam on Second Harmonic Generation in collisional Magneto-plasma

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Impact of high-power Cosh-Gaussian laser beam on second harmonic generation in collisional Magneto-plasma is investigated in this communication. When a laser beam transitions along an external magnetic field, it can propagate in two different ways in collisional magnetized plasmas: ordinary and extra-ordinary modes. The change in magnetic field causes the redistribution of electrons. The non-uniform heating causes density gradients to form in the transverse direction of the pump beam. Density gradients cause the electron plasma wave (EPW) to be stimulated at the pump beam frequency. EPW and pump wave interact nonlinearly to generate second harmonics. A well-established paraxial theory approach has been used to derive the 2nd order ODE for the pump beam's beam waist and the efficiency of the second harmonics. Numerical results have been obtained for the differential equation solution and the efficiency of the second harmonics. Lastly, the impact of external magnetic fields and well-established laser-plasma parameters on the pump beam's waist and second harmonic yield are also examined. As these results are useful for laser driven fusion experiments which can reduce the plasma Instabilities.

Application and damages of W/Cu flat-type components used as lower divertor target in EAST

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The W/Cu flat-type component, which has flexible cooling structures and thus hopes to withstand and exhaust high steady state heat fluxes up to 20MWm-2 or even more, is a suitable candidate plasma facing component (PFC) for the divertor in future fusion devices. Based on current explosively welding and brazing technologies, a kind of high performance W/Cu flat-type component with hyper-vapotron heat sink structure was developed, which passes through the harsh high heat load tests (20 MWm-2,1000 cycles). Thus, it was thus installed and tested for the divertor target in the EAST, to identify its reliability and capacity under actual tokamka plasma conditions.
The W/Cu flat-type components were designed with symmetric bevels at both sides to counteract the leading edge effect, and mainly installed as the outer horizontal targets to withstand the high heat load. In recent years, during the long-pulse and high-power plasma campaigns, most W/Cu flat-type components for divertor target often show good performance with an acceptable maximum temperature only about hundreds oC in generally in the early plasma discharges during each plasma campaign, and thus promote the EAST to successfully achieve many high performance plasma discharges (#98958, #106915 and #122296). However, after a period of cyclic plasma discharges, some W/Cu flat-type components begin to show some abnormal phenomenon, with observation of bright dots during plasma discharge and residual dots after plasma quench. What is more, a large number of droplets ejected from the bright dots could be happen occasionally, which were due to the melting of the W/Cu flat-type PFC. Post mortem inspection after each plasma campaign also shows severe damages in form of cracking, melting, and even exfoliation of the W plate on flat-type components. First of all, the features of such damages are characterized and analyzed. The macro cracks on W plates were often along toroidal or poloidal direction, and always propagated cross the whole plate and thus divided the single W plate into two or more isolated parts. Meanwhile, the cracks can be also found at the joint interface by means of both ultrasonic detection and microscopic observation. The propagation of the cracks at joint interface cause the deterioration the heat removal ability and thus resulted in an increased peak temperature at W plates. Repetition of such process gradually destroyed the structure of the W/Cu flat-type components and decreased the performance until the melting and exfoliation occurred. The formation mechanism of cracks were analyzed by means of thermal stress and strain distribution and evolution simulated by finite element method (ANSYS).
The W/Cu flat-type PFCs for divertor target in EAST is an attempt to apply this PFC in actual tokamak conditions, which provides unique reference and key lesson for future fusion devices.

Impact of Long Outer Legs on Divertor Power Distribution in Single-Null Power-Plant-Scale Tokamaks

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Accurate prediction of heat and particle loads on divertor plates is critical for the successful design and operation of tokamaks. As devices approach power-plant-relevant conditions, it becomes increasingly difficult to avoid exceeding material and engineering limits of the plasma facing components. An experimentally-motivated assumption widely used in 0-dimensional modeling is that the outer target receives the majority of the exhaust power [1]. This is beneficial, as the outer target is at a larger major radius than the inner one, and thus has a wider surface area, leading to lower peak heat fluxes and particle flow densities. Maintaining this assumption, long outer legs and advanced magnetic topologies have been proposed as potential solutions for power plant power exhaust in Single-Null (SN) configurations [2]. However, results from SPARC SOLPS-ITER L-mode modeling highlight potential drawbacks associated with longer outer legs: over a scan of upstream density (Greenwald fraction = 0.13–0.22) and power (Psol = 4–15 MW), longer-legged scenarios yield more than 60% of power flowing to the inner target, with peak unmitigated heat flux densities up to 40 MW/m², more than an order of magnitude greater than on the outer target.

Dependency of in-out power ratio on upstream collisionality and divertor leg lengths is observed and compared with analytical scalings [3], showing that increasing the outer leg length in power-plant-relevant conditions leads to an increase in power flowing inboard. This trend is observed across both simple long-legged divertors and advanced geometries, such as X-Point-Target and Super-X, as well as various target plate orientations, further suggesting that the enhanced parallel distance between the stagnation point and the outer divertor plate is the primary driver of these asymmetries.

Characterizing these asymmetries is critical for informing divertor design and operational strategies to manage extreme heat fluxes in future machines. Developing reduced models to quantify in-out power sharing from geometrical and plasma boundary parameters would thus be beneficial to refine modeling with 0-dimensional scoping tools. These findings suggest that while long outer legs can reduce outer target heat loads, their implementation in SN configurations may lead to unintended power redistribution to the inner target: this highlights the need to combine long-leg divertors with double-null operation or other mitigation strategies to ensure balanced power distribution.

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Progress of neutral beam injection on ST40 Tokamak

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The ST40 is a compact high field tokamak currently operating at Tokamak Energy in the UK. This experimental machine has achieved outstanding plasma performance in recent years, expanding our knowledge of plasma behaviour in spherical tokamaks. Presently ST40 has two neutral beam injector systems installed that regularly provide up to 1.8MW heating on plasma pulses. Both beams have been used extensively in ST40 experimental campaigns to provide heating, hot ion fuelling, plasma stabilisation and current drive. They are also both individually used for charge exchange recombination spectroscopy, an important diagnostic used in the analysis of the plasma.
The continued operation of the neutral beam injectors has greatly expanded our knowledge and understanding of these systems and allowed optimisations and upgrades to be made across both NBIs. We will discuss the NBI results from experimental campaigns and talk through the steps that have been taken to improve the performance and reliability of each beam system, through changes to operational parameters and upgrades to auxiliary systems. Both NBIs will be discussed separately due to differences in technical set up, operation and uses in plasma experiments, with the differences being individually highlighted and discussed.

Integrated Diagnostics and Hardware Design of a Cryogenic Pellet Ablation Test Stand

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A fusion technology platform dedicated to developing a novel cryogenic pellet ablation test stand is under construction at Columbia University in collaboration with Oak Ridge National Laboratory to support Disruption Mitigation Systems (DMS) worldwide by providing the first direct ablation measurements with controlled incident particle energy, yielding more direct data for tokamak disruption mitigation and fueling models. The platform employs a pipe gun approach for pellet formation and injection, where gas is fed into a cold zone maintained below the triple-point temperature to enable controlled desublimation. Once solid, a cooled copper pneumatic punch launches the pellet into the ablation chamber, where an optical sensor activates a high-speed camera and electron beam, partially ablating the pellet. Fractional mass loss and speed from particle ablation are determined using microwave diagnostics, high-speed imaging, and a Python-based pellet tracking system. In October, the first H_2 pellet was successfully produced. This contribution highlights hardware innovations, diagnostic advancements, and system integration within the test stand.

*Work supported by US DOE and Columbia University

Vacuum Pressure Impregnation of the Toroidal Field Coil for the NSTX-Upgrade Fusion Device

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The National Spherical Torus eXperiment (NSTX) has undergone a major upgrade to NSTX-U at Princeton Plasma Physics Laboratory (PPPL). NSTX-U will double the toroidal field, plasma current, and neutral beam injection heating power, as well as significantly increase the pulse duration. Currently NSTX-U is undergoing a recovery effort to restart experiments. One of the critical jobs included in the recovery effort is to replace the legacy toroidal field/ohmic heating (TF/OH) magnet bundle. The TF bundle is at the center of the TF/OH bundle, which is a thirty-six (36) turn copper coil bundle that forms the inner legs of the toroidal field coil system. It operates at a voltage of 1013V and a current of 129,778 Amps. In this paper, the TF bundle insulation system design will be covered. Specimens were developed to validate the mechanical and electrical properties with S2 fiber glass and the CTD 425 resin insulation system. The vacuum pressure impregnation (VPI) method was also developed and used for the specimen fabrication. S2 glass fabrication process will be briefly covered, and glass compaction tests were performed to find out the compression ratio and pressure relationship, as well as the modulus of the S2 glass insulation, which then was used in the TF conduction stacking and compression analysis to make sure proper S2 compaction ratio was achieved within the TF bundle. The detailed TF conductor compression method will be presented, as well as the jig and VPI mould design. Four quadrants with 9 conductors compacted in each quadrant have been successfully VPIed with good VPI quality. The mechanical and chemical analysis result for the VPIed resin system, as well as the electrical tests and geometrical metrology result of the VPIed TF quadrants will be covered in detail at the end of the paper.

Assessment of delamination behavior of joint between tungsten and copper interlayer within divertor target monoblock by parametric studies of cohesive zone model

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JA-DEMO refers the structure design of the ITER divertor. To protect divertor from enormous heat flux of plasma atmosphere, tungsten monoblocks are arranged into the surface of the divertor. Besides, cooling pipe is jointed with copper interlayer to remove heat from the surface. Because of a large gap of the thermal expansion coefficient, delamination can happen. This phenomenon is of serious concern in that it poses the threat of surface overheating and total failure. To assess the possibility and behavior of delamination, the authors have proposed utilizing finite element analysis applying the cohesive zone model. This method needs to input fracture mechanics parameters of interface such as adhesion strength and critical energy release rate, while the parameters of the tungsten-copper interface are still unknown and difficult to be experimentally evaluated particularly for high temperature. In this study, therefore, parametric studies on these mechanical properties are conducted to elucidate the fracture behavior of the joint interface, particularly dominant crack propagation mode. A half cut finite element analysis model of monoblock including the copper interlayer and cooling tube is built. 20 MW/m2 heat flux is applied to the surface of tungsten monoblock for thermal and structure coupling analysis. Water cooling temperature and pressure is set to be 120 ℃ and 3.3 MPa respectively. Representative relation between crack stress and opening displacement (traction-separation law) is assumed to be bilinear. This relation is expressed only by the maximum traction and critical energy release rate. Parametric studies on these parameters of the tungsten-copper interface are conducted considering weaker bulk material and its temperature dependency. Additionally, delaminated or sticked discrimination maps regarding the relation between the maximum traction and the critical energy release rate (or the maximum separation displacement) are summarized. In all cases, the joint interface was closed by compression during heating. This result excludes the possibility of mode Ⅰ (normal tensile crack opening) delamination. In the cases of delamination, applied interfacial shear stress and displacement exceeded their threshold. Therefore, it is suggested that the mode Ⅱ (shear crack opening) is a dominant factor for the delamination. Quantitative delamination criterions were provided. Delamination was not predicted in the case that exceeds 100 MPa of the maximum traction and 0.2 mm of the maximum separation displacement, where the critical energy release rate exceeded 0.5 kJ/m2. Because there is no experimental data of the interfacial mechanical parameters, the possibility of delamination will be evaluated and discussed by estimating each mechanical parameter of interface and referring the delaminated or sticked discrimination maps, considering the decrease of interfacial to bulk strength ratio at room temperature and temperature dependency of bulk strength reported in previous studies.

Advancements in the fabrication of tungsten-fiber reinforced copper composite tubes for plasma facing components

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The development of high-performance materials for plasma facing components is a key point for the development of fusion reactors. Among the envisioned concepts, the tungsten-fiber-reinforced copper (Wf-Cu) composite tubes represent an innovative solution for the EU-DEMO Divertor target, addressing the limitations of conventional ITER-like CuCrZr pipes. In fact, these composite tubes provide enhanced mechanical resistance, less affected by degradation caused to high temperatures and neutron damage respect the CuCrZr pipes, as well as reduced amount of radioactive waste due to less W activation respect to the copper. Currently the two procedures under investigation for the fabrication of Wf-Cu composite tubes are the copper casting and the galvanic process. Furthermore, a novel study introduces a copper impregnation technique to fabricate Wf-Cu composite tubes by joining pre-wrapped tungsten fibers and tungsten monoblocks in a single process, eliminating the need for brazing alloys and their associated issue of neutron activation. The research aims at the fabrication of tubes suitable with small and medium scale mock-up manufacturing, assessing their compatibility with EU-DEMO divertor target operating conditions. In the present paper, the advancements in the Wf-Cu composite pipes manufacturing are summarized. The fabrication processes described underwent several optimizations, including modifications in the W fiber density and weaving structure, development of refined tools and optimized procedures to achieve enhancements in copper infiltration quality, with reduced defect size and number, as well as in uniformity of the internal and external tube surfaces. The mechanical integrity of the composite tubes has been assessed throughout the integration of non-destructive (e.g., ultrasonic testing) and destructive (e.g., metallographic) examination techniques. X-ray tomography as been also used as an alternative non-destructive examination method to evaluate the presence potential internal defects within the Cu matrix, providing a comprehensive understanding of the structural integrity of the samples. Furthermore, the highest quality samples are used to perform a mechanical characterization at high temperatures, along with microstructural and fractographic analyses to evaluate the behavior of Wf-Cu tubes under DEMO-relevant operating conditions.

Presentation of the assembly and commissioning of a representational set of high filed HTS magnets in a reactor relevant configuration

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• Tokamak Energy is pursuing commercial fusion energy based on the development of spherical tokamaks with high temperature superconducting (HTS) magnets.
• The Demo4 project is an ambitious high field HTS magnet build that has delivered a full spherical tokamak magnet system. This will provide a world first demonstration of the operation of a representative array of coils in a toroidal and poloidal configuration, operating at fusion-relevant magnetic fields and temperatures.
• This presentation will provide a system-level overview of the magnet system and its key components, including HTS coils, vacuum vessel, bespoke cryogenic cooling system, mechanical structure, instrumentation and other systems.
• 2024 saw the accelerated build of Demo4, taking 28 Toroidal field coils, and 16 Poloidal field coils and building them into a cold mass assembly with over 600 sensors. Phase testing of sub-assemblies down to 18K was completed, testing both the power supplies and TF current leads up to full energisation current. Early 2025 saw the full HTS cold mass suspended below the cryostat top plate and enter into the commissioning phase.
• The assembly sequence will be shown from coil winding to commissioning at 20K
• The sequence of initial cool down and commissioning, leading to the energisation of the TF and PF HTS coils will be presented.

Engineering design of the thermal shield for the Divertor Tokamak Test (DTT) facility

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One of the most critical challenges in magnetic confinement fusion is the management of large heat and particle fluxes. To address this issue, the Divertor Tokamak Test (DTT) facility is currently under construction at ENEA Frascati premises. It will test different power exhaust solutions for the EU DEMO reactor. Different plasma configurations will be required, asking for a significant flexibility and reliability of the different subsystems.

DTT will be a fully superconducting tokamak adopting the state-of-art low critical temperature cable in conduit superconductors. The poloidal and toroidal field coils, and the central solenoid, will be operated at a temperature of 4.3 K, being cooled by supercritical He. As the magnets will face warm and hot components as the vacuum vessel, its ports and the cryostat, a suitable thermal shield is required to reduce the heat fluxes on the coils.

In this work, the engineering design of the thermal shield will be presented. It will be composed by a set of double-walled stainless steel panels surrounding the vacuum vessel (VV) and its port extensions, while single-walled panels with multi-layer insulation will be used for the cryostat. It will have to fit in the very small space between the magnets and the VV, guaranteeing no interference with the two in any operative condition.

From the structural point of view, the thermal shield will have to sustain both thermo-mechanical loads, due to its weight and to the thermal contraction from room to its operating temperature, and electro-magnetic loads, due to the varying magnetic field. From the thermal point of view, it will face the radiative heat load from the hot components, and the conductive heat load from its gravity supports. The solutions adopted to satisfy these thermal and mechanical requirements will be presented, e.g. the use of gaseous He at ~20 bar and ~80 K to actively cool the surface of the panels, or the surface polishing to reduce the emissivity and consequently increase the reflectivity.

Progress in the realization of solid-state ICRH transmitters for DTT

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Solid-state power amplifiers (SSPAs) with an unprecedented combination of bandwidth, output power, and pulse length are being developed for the Ion Cyclotron Resonance Heating (ICRH) system of the Divertor Tokamak Test facility (DTT). The unavailability of suitable off-the-shelf grid tubes as well as cavities in the ion cyclotron range of frequencies selected for DTT, together with the impressive advances of SSPAs, e.g. in the field of accelerator physics, drove the choice of the technology for the ICRH transmitters of DTT towards the use of solid-state devices. The procurement of the first two transmitters is now ongoing and their detailed design was recently frozen. They will provide a nominal output power of 1.2 MW with VSWR=1.5, for 50 s every hour, over the frequency range from 60 to 90 MHz. This paper provides an overview of chosen transmitter architecture and reports the test results of the first radiofrequency (RF) module that constitutes the main building block of the SSPA.
Each ICRH transmitter of DTT will consist of 128 RF modules that have been conceived adapting to ICRH needs a robust design reliably used in broadcasting transmitters. In the latter industry, Laterally-Diffused Metal-Oxide Semiconductors (LDMOS) have been used as active device since many years. The RF module of DTT transmitters relies on twelve printed circuit boards, called RF pallets, where a transistor of this type is mounted. DC voltages for transistor polarization will be provided by more than 300 off-the-shelf power supplies, external to RF modules and arranged in a parallel configuration to improve fault tolerance. A central control system orchestrates transmitter operation, interacting with locally distributed logic modules and with the low-level RF unit, which is also part of the procurement.
As far as the combination of all 128 RF modules are concerned, the frequency range required for DTT, with a fractional bandwidth of 0.4, prevents from relying on a cavity resonator like those developed and employed for single-frequency applications. A gradual, multi-stage, combination strategy has been thus designed, optimizing the performance at the edges of the frequency range, where the transmitters are mostly expected to operate.

Fusion Innovative Research Opportunities on Maturing Superconducting Magnet Technology

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The past decade has seen a rapid proliferation in the numbers and varieties of commercial fusion power start-ups, all vying to put electrical power on the grid by the middle of the century. The demand for qualified magnetic fusion technology personnel has grown so rapidly that it far outpaces the development of the workforce needed to support this nascent industry. This has led several start-ups to partner with existing, DOE-supported fusion laboratories to access the expertise required to achieve their goals. Significant Innovation Research opportunities still exist in maturing Fusion Superconducting Magnet Technologies as a broad-based means to formally address existing gaps in the commercial fusion realm. Basic fusion magnet technologies applicable to a wide range of fusion devices, not just one device or concept are needed, to make those freely available to the community. Technology for high-field magnetic confinement fusion devices has thus far focused almost entirely on REBCO. While REBCO is a marvelous conductor with extremely high current density at high magnetic fields, it is not its cartoon and it is not ideal for all applications, not the least because of its high cost and complex processing. REBCO is available only in tape form, resulting in high magnetization (screening current) loss in rapidly changing magnetic fields as well as highly non-uniform stress distribution. Not every fusion concept needs the high-field capability of REBCO. Several fusion start-ups we've engaged are pursuing lower-field, lower-cost alternatives for which MgB2, Bi-2212 or even Nb3Sn could potentially be better options if sufficient information regarding suitability were more widely available. Based on feedback from industry we intend to advance not only high field REBCO but also these promising, but lesser-studied options as part of our developmental activities.

SECOND-HARMONIC GENERATION OF INTENSE LAGUERRE GAUSSIAN LASER BEAM IN RELATIVISTIC-PONDEROMOTIVE PLASMA

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Self-focusing and second harmonic generation of Laguerre Gaussian laser beam have been investigated under the collective effect of relativistic-ponderomotive nonlinearities in plasma with an upward density transition. For the high-intensity gradient of the laser beam, ponderomotive nonlinearity becomes important. However, at a much higher intensity if the oscillatory plasma electron velocity becomes closer to the speed of light, the role of relativistic nonlinearity also becomes important. The collective effect of both causes a transverse density gradient that excites an electron plasma wave inside the plasma, which has the same frequency as that of the incident laser beam. When excited and coupled with the fundamental laser beam, the plasma waves yield second harmonic radiation. WKB approximations and the moment theory approach have been used to derive an expression for the laser beam width and the corresponding second harmonic yield. An enhancement in self-focusing and second harmonic yield has been seen due to the inclusion of density transition. The results of the current analysis have led to the conclusion that the self-focusing and harmonic yield of the laser beam can be enhanced by utilizing the collective effect of the plasma density ramp and relativistic-ponderomotive nonlinearities in plasma along with higher modes of the Laguerre-Gaussian laser beam.
The self-focusing of the laser beam concentrates laser energy on the target, thereby enhancing the efficiency of energy deposition into the fuel pellet. The increased intensity due to self-focusing facilitates improved compression of the fuel, achieving the high temperatures and pressures essential for Inertial Confinement Fusion (ICF). Additionally, a higher second harmonic yield enhances the optical conversion efficiency of laser energy into the fuel pellet.
In summary, these effects collectively optimize the conditions for ICF, increasing the likelihood of its successful implementation. The findings of the present analysis will serve as a valuable resource for researchers working in the field of ICF.

Plasma Characterization Studies Between 4-8 MHz with the SupRISE Test Device for the DIII-D Neutral Beam Injection System Upgrade

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To further support the development of inductively coupled plasma (ICP) sources for the DIII-D Neutral Beam Injection (NBI) system, a full-scale ICP test device has been constructed at the DIII-D facility. The Superior Radiofrequency Ion Source Experiment (SupRISE) test device aims to investigate how power coupling to the plasma is impacted by drive frequency and chamber dimensions. SupRISE uses a four-turn copper antenna to couple up to 50 kW of RF power through a racetrack shaped (~70 x 30 x 30 cm) quartz vessel at frequencies between 4-8 MHz. A 3 mm thick Faraday shield lines the inner walls of the dielectric to minimize sputtering from high energy ion bombardment. With the use of novel solid state RF generator technology,

SupRISE is capable of achieving plasma densities of ~$10^{18}/m^3$ with a duty cycle of 10 s ON and 210 s OFF as required during DIII-D NBI operations.

High plasma uniformity is required for any neutral beam ion source to minimize beam divergence. Excessive divergence reduces the total injected power into the DIII-D tokamak and can significantly damage beamline components. A plasma diagnostic has been developed and used in SupRISE to measure the density profile across the extraction aperture. The density

profiles measured with this spatial probe array under various operational conditions will be presented.

Significant research has been devoted to large volume ICPs using drive frequencies of 1 or 2 MHz. However, the 4-8 MHz range at high powers remains relatively unexplored experimentally. This can be attributed to the expected increase in Ohmic heating losses at higher frequencies. Experiments on SupRISE have been conducted to fully characterize how coupling efficiency depends on drive frequency over this range. Results of this frequency study will be presented with and without the presence of a Faraday shield. This research and parallel efforts at our partner institution (North Carolina State University) are essential for the development of ICP sources at the DIII-D National Fusion Facility.

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Deploying MARTe2 and MDSplus for Scalable Real-Time Control Systems: A Proof-of-Concept for the SMART Tokamak

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The Princeton Plasma Physics Laboratory (PPPL) is collaborating with the University of Seville to implement a proof-of-concept deployment of the MARTe2 software framework integrated with MDSplus to support the control system of the SMall Aspect Ratio Tokamak (SMART). This effort aims to demonstrate the feasibility and advantages of a modular and scalable control architecture for real-time tokamak operations, leveraging PPPL's advanced infrastructure and expertise in fusion energy research.

MARTe2 is a C++ modular and multi-platform framework designed specifically for developing real-time control system applications. Building on the success of its predecessor, MARTe — which was extensively deployed in fusion experiments like the Joint European Torus (JET) — MARTe2 introduces a robust software Quality Assurance (QA) strategy and improved modularity. The framework's architecture emphasizes the separation of platform-specific implementations, environmental details, and user algorithms, enabling the reuse of components across diverse environments and seamless transitions between development and deployment stages.

MDSplus, a widely used data management system in the fusion community, complements MARTe2 by providing tools for experiment data storage, retrieval, and configuration. Its integration ensures streamlined management of large-scale data generated during tokamak discharges, supporting both real-time and offline analysis.

The SMART tokamak is a spherical tokamak designed to explore advanced plasma confinement configurations and to investigate the potential of negative triangularity as a means to stabilize plasmas. This collaboration provides an ideal testbed for deploying MARTe2 and MDSplus, validating their performance in a cutting-edge research environment.

This work also provides a stepping stone to integrate MARTe as an additional framework to use for Instrumentation and Control, diagnostic, and control related systems at PPPL, leveraging the benefits of a tested infrastructure over custom one-off implementations for future systems.
This work outlines the design and implementation strategy for integrating MARTe2 and MDSplus into SMART's control system, including the challenges addressed and the expected benefits of this approach. Key goals include achieving deterministic real-time control, enhancing data accessibility, and ensuring the system's scalability to support future experimental upgrades. The results of this proof-of-concept deployment will provide critical insights into the suitability of MARTe2 and MDSplus as foundational tools for next-generation tokamak control systems, contributing to broader efforts in advancing fusion energy research.

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Dynamic compensation of plasma-coupled voltage for EAST quench detection based on the feedback of plasma operation parameters

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The evolution of key plasma parameters during full superconducting tokamak operation produces coupled voltage to the quench detection signal, which directly influences some critical issues, such as the reliability of quench detection and the security of superconducting magnet. The studies found that the plasma-coupled inductance is time-varying owing to plasma's ever-changing shapes and other key parameters, and the conventional decoupling and compensating methods cannot effectively suppress the plasma-coupled voltage. This paper investigates the coupling mechanism and suppression strategy under different plasma discharge scenarios in the experimental advanced superconducting tokamak (EAST), establishes the inversion model of dynamic compensation coefficient based on the feedback of plasma operation parameters. Through off-line computation and real-time inversion, a novel dynamic compensation strategy of plasma-coupled voltage for EAST quench detection was developed to break the limitations of fixed coefficient on the compensation effectiveness, and achieve the higher compensation precision of the plasma-coupled voltage. The experimental results indicate that this novel method can suppress the residual plasma-coupled voltages in EAST PF and TF quench detection signals within 50 mV and 10 mV, respectively, to further improve the Signal-to-Noise Ratio (SNR) and reliability of EAST quench detection system, which provides robust assurance for the secure and stable operation of ITER and future full superconducting fusion reactors.

Commissioning the JT-60SA VUV Divertor Spectrometer

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Wednesday Posters 1, Lobdell (Building W20 Room 208), June 25, 2025, 10:30 AM - 12:00 PM

This work presents the final commissioning steps of the Vacuum Ultraviolet (VUV) Spectrometer for the JT-60SA divertor, specifically focusing on completing its assembly, alignment and calibration prior to the installation of the instrument on the machine. The diagnostic system is designed to assess impurity contributions to the radiation emitted from the divertor region in various scenarios and to study divertor physics, including plasma detachment, in support of JT-60SA's mission to advance high-β steady state operations and support ITER exploitation.
The spectrometer is a double SPRED system mounted on a custom chassis at an upper port of the JT-60SA vessel to have a clear view of the divertor region. It features a support structure outside the cryostat and a port plug inside the tokamak, housing a remotely controllable optical system with gold-coated toroidal mirrors to direct the VUV emission to the spectrometer. This work details the system's mechanical design, installation, and optical chain alignment procedure.
The instrument simultaneously acquires spectra in the 10–48 nm and 44–125 nm ranges, with resolutions of 0.08 nm and 0.14 nm, respectively. Dispersed light is captured by high-sensitivity CCD detectors. The advanced toroidal gratings correct aberrations, enhance wavelength resolution, and enable 1D imaging detection, allowing for spectral resolution along divertor radial direction with ideal spatial resolution of 100 mm. Additionally, this work details the gratings alignment procedure and the tests carried out to characterize the imaging capability of the optical system. Finally, the calibration functions for both channels are obtained.

Parametric and Monte Carlo design tool to manage uncertainty in the concept design of STEP toroidal field coils and remountable magnet joints

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Wednesday Posters 1, Lobdell (Building W20 Room 208), June 25, 2025, 10:30 AM - 12:00 PM

In the UK's Spherical Tokamak for Energy Production (STEP) the toroidal field (TF) coils have a significant influence on the inboard radius of the tokamak machine, which largely determines the overall size. The design of high-temperature superconductor (HTS) TF coils must manage considerable uncertainty, due to the limited maturity of the technology. STEP incorporates remountable magnet joints, enabling the upper section and centre column of the TF coil to be removed for vertical maintenance access. However, these joints contribute additional complexity and uncertainty to the overall TF design. This study presents a parametric modelling and Monte Carlo analysis tool to quantify and manage design uncertainty during the conceptual design of the STEP TF coil. The tool enables engineers to make informed decisions by evaluating the impact of design changes on the inboard radius of the machine. A parametric tool was developed in Microsoft Excel which enables users to define nominal input parameters that drive the concept design geometry. Key inputs, such as winding current, toroidal field strength, and plasma radius can be adjusted by the user. The tool calculates the nominal TF coil geometry and optimises the design by identifying a configuration that minimises inboard radius. Users can define parameter ranges with upper and lower limits. A Python script performs sensitivity analysis within these ranges, with results presented as a tornado plot that quantifies the impact of each parameter on inboard radius. The user may assign probability distributions to an input parameter range, reflecting the uncertainty. With this, a second Python script performs Monte Carlo analysis to provide a probabilistic assessment of the inboard radius. The tool has been key to developing the STEP concept baseline design, successfully producing optimised nominal TF coil geometry from defined inputs. Tornado plots from sensitivity analysis identified key size driving parameters such as the cooling channel diameter, highlighting opportunities to reduce inboard radius. Monte Carlo analysis provided statistical confidence in the inboard radius of the machine. The tool has demonstrated a powerful framework for managing design uncertainty in the development of TF coil magnets, enabling an informed decision on the STEP concept machine size.

3D impurity transport modeling assessment of the effect of OSP location and Bx▽B drift direction on W erosion and migration in the DIII-D SAS-VW divertor

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Wednesday Posters 1, Lobdell (Building W20 Room 208), June 25, 2025, 10:30 AM - 12:00 PM

Tungsten (W) erosion and migration from the DIII-D Small Angle Slot V-shaped W-coated (SAS-VW) divertor were analyzed using the 3D Monte Carlo transport code GITR and compared with experimental diagnostic measurements performed by optical spectroscopy. Four scenarios were investigated, with two outer strikepoint (OSP) locations and both Bx▽B drift directions.

For a DIII-D discharge with the Bx▽B drift oriented towards the divertor and the OSP on the outboard (W-coated) side of the slot, W gross erosion is predicted to be ~1e19 m-2s-1, an order of magnitude greater than estimated by the experimental optical emission spectroscopic W gross erosion. The ExB drift drives W ions from the near-SOL into the private flux region, where friction dominates and pushes them deeper into the slot, resulting in ~99.9% of W depositing in the divertor region.

With the same Bx▽B drift direction, but the OSP located at the vertex of the slot, the ion and electron temperatures increase more steadily along the W surface from the vertex to the outer slot opening. W gross erosion follows a similar trend, although with an order of magnitude smaller value of ~1e18 m-2s-1. With the private plasma farther from the W surface, the ExB drift is not strong enough for as many of the sputtered W ions to cross the separatrix.

For both strikepoint locations with the Bx▽B drift oriented towards the divertor, attached plasma conditions were achieved, leading to larger W gross erosion than the conditions with Bx▽B drift oriented away from the divertor. More than 80% of the redeposition is prompt and the remainder is entrained redeposition, dominated by a balance between the ExB drift, friction force, and temperature gradient force.

The remaining two scenarios consider both OSP locations, but with a Bx▽B drift oriented away from the divertor. Both scenarios achieved detached divertor conditions with predicted plasma boundary temperatures below the sputtering threshold for W using a simple sheath model. Because of this, a particle-in-cell sheath simulation was performed using the hPIC2 code for 110 points along the discretized W divertor surface for all four scenarios. The inclusion of this higher-fidelity physics sheath model predicts a high-energy tail of incident D and C species that results in predictions of W gross erosion. As expected from the detached conditions, the magnitude of W gross erosion for these cases is closer to ~1e14 m-2s-1. Additionally, the W ionization mean free path is longer, leading to a higher W neutral population and less prompt redeposition.

The balance of W gross erosion and redeposition in the divertor contribute to W leakage out of the divertor, which will be reported for all scenarios. A permutative force analysis will also be described, revealing the dominating force and drift contributions in each scenario.

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Research on characteristics of high power RF amplifier based on two-port network model

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Wednesday Posters 1, Lobdell (Building W20 Room 208), June 25, 2025, 10:30 AM - 12:00 PM

High power RF amplifiers are an important part of ion cyclotron resonance frequency systems (ICRF), which play a crucial role in transmitting RF energy in plasma heating. In this paper, based on the two-port network model, the theoretical calculation of high power RF amplifier is carried out, which mainly includes the distribution of anode power dissipation and anode efficiency with the output power when the anode output resistance, voltage standing wave ratio (VSWR), anode voltage and tetrode conduction Angle change. By comparing with the experimental results, the changes of theoretical calculation and experimental results are highly consistent, which verifies the validity of the model. The findings provide valuable insights for optimizing the design and performance of RF amplifiers, especially in megawatt-level power applications, and help to develop more reliable and efficient high-power RF amplifiers.

Utilizing Lattice-Based Support Structures to Improve Part Accuracy in Laser Powder Bed Fusion Additively Manufactured GRCop-42 Lower Hybrid Current Drive Launchers

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Additive manufacturing (AM) using laser powder bed fusion (LPBF) provides design flexibility for fabricating complex geometries unachievable utilizing traditional manufacturing techniques, such as Lower Hybrid Current Drive (LHCD) Radio Frequency (RF) launchers used in DIII-D. Part distortion and build plate warpage are known issues caused by rapid thermal cycles and high thermal stresses inherent in the build process, risking build failure. Two monolithic LHCD launchers were manufactured with LPBF using Glenn Research Copper 42 (GRCop-42), a Niobium Chromide (Cr2Nb) 8 at. % Cr, 4 at. % Nb precipitation hardened alloy on Carbon Steel (carbon wt. 0.45 %) build plates with and without an underlying geometric lattice-based support structure. A parametric design study was first performed using stereolithography (SLA) additive manufacturing developed tree-like support structures and explored the effects of varying support parameters in mitigating the curvature of a detachable build plate. The heights of each build plate were profiled. Results of the study show a significant reduction of build plate curvature by up to 68% utilizing raft support structures. Similar improvements were demonstrated in LPBF manufacturing of the LHCD launchers, reducing curvature by 36% compared to an unsupported build. Measurements of the internal aperture heights after the build process also demonstrate a substantial 40% improvement in precision with the support structure. A hot isostatic pressing (HIP) step was performed to consolidate residual internal voids in the LPBF process and relieved residual stresses and part distortion. After this HIP step, the launcher utilizing support structures had a 66% improvement in aperture heights. These results demonstrate the effectiveness of raft tree-like support structures for enhancing L-PBF manufacturing performance.

Calibrated Miter Bend Polarimeter

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Wednesday Posters 1, Lobdell (Building W20 Room 208), June 25, 2025, 10:30 AM - 12:00 PM

Overmoded microwave waveguides are commonly used in fusion for low loss microwave transmission in heating and current drive (electron cyclotron heating) and diagnostics (electron cyclotron emission, reflectometry, collective Thomson scattering). Such systems require protection from high power microwaves, making it important to monitor the signals being transmitted or received. Power and polarization in particular are two key parameters to measure in transmission lines. Because of the quasi-optic nature of these waveguides, it is challenging to design robust polarimeters that can measure instantaneous microwave power and polarization. One common power monitor design uses an array of coupling holes in a miter bend mirror, with a directional coupler auxiliary waveguide placed behind the mirror. This design, however, typically only couples into one polarization. Dual polarization detection requires either two miter bends, or two rows of coupling holes in the same miter bend mirror. The former option adds unneccessary mode conversion and signal correlation, and the latter option is suceptible to error from offset beams since both rows of holes cannot occupy the same center in the mirror. This approach also adds complexity since two coupling channels must be machined. In this work, a single-channel polarimeter miter bend is proposed that uses a circular auxiliary waveguide, instead of a conventional rectangular one. This design can detect both polarizations simultaneously with just one row of coupling holes, and two coupling ports (forward and reverse). A conventional orthomode transducer can then separate the two polarizations into separate power detectors. This essentially allows for the power monitor miter bend to provide complete polarization information in both the forward and reverse directions. The design is beneficial for stray radiation monitors to protect diagnostics when the polarization of the stray radiation detector is not known. It can easily be extended to high power, where knowledge of the polarization of electron cyclotron heating signals launched into the plasma is essential.
This work is supported by General Atomics Corporate funding.

Using Magnum-PSI to replicate fusion plasma and transient exposure on Eurofer-97

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Wednesday Posters 1, Lobdell (Building W20 Room 208), June 25, 2025, 10:30 AM - 12:00 PM

Inside a future fusion tokamak, the high temperatures of the plasma can present challenges to the structural material of the tokamak walls. Alongside the steady state plasma exposure, structural material in a tokamak fusion reactor is also expected to receive up to 2 GW/m² of heat flux during plasma disruptions [1]. These events are predicted to occur over 2500 times during a DEMO-style tokamak's lifetime [2], raising the material's temperature to around 950˚C and vaporising the coolant. Currently, the leading candidate for tokamak structural material is the reduced activation ferritic martensitic steel, Eurofer-97. Previous work with a CO_2 laser has demonstrated that these transient events cause a significant microstructural transformation, replacing Eurofer-97's martensitic laths with a ferritic structure, including grains that are nine times the size of the grains present in the as-received sample. Transients also encourage the growth of both M_7C_3 and $M_{23}C_6$ precipitates, decreasing hardness by 32% and affecting its magnetic properties [3]. Although they demonstrated the importance of understanding thermal transients, these previous experiments were limited by heating and cooling rates significantly slower than the transient durations predicted in tokamaks.

To remedy this, specimens of Eurofer-97 were taken to the Magnum-PSI facility in Eindhoven and were exposed to steady state plasma and laser transients simultaneously to better replicate fusion operating conditions. The material was subjected to a combination of 500˚C argon plasma and laser pulses with a ΔT of 450-500˚C. These pulses ranged from 0.5-3 ms in duration and were between 1 and 1000 in number. Results from the most extreme exposure of 1000 laser pulses of 3 ms duration found large surface globule formation, with a maximum globule diameter of 146 µm. Melted droplets were also found in the surface cracks which energy dispersive x-ray (EDX) analysis revealed to be melted bulk material. Cross sectional electron backscatter diffraction (EBSD) analysis found a region of recrystallised grains under the surface of the sample, with the transient region experiencing a smaller grain size than the sole plasma exposure. This is most likely due to the rapid heating and cooling experienced. Further comparative cross sectional EBSD will be presented, alongside cross-sectional hardness testing and quantitative measurements. The implication of these results on fusion structural materials will also be discussed.

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Progress on Fusion Magnet Multiphysics Simulation using the Open-Source MOOSE Framework

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Wednesday Posters 1, Lobdell (Building W20 Room 208), June 25, 2025, 10:30 AM - 12:00 PM

In a magnetic confinement fusion (MCF) device such as a tokamak, toroidal and poloidal superconducting magnets confine and shape the plasma to promote the fusion reaction. Some designs being considered by the industry for MCF devices have high magnetic field requirements, upwards of 10 Tesla. This leads to high amounts of stored energy in the magnets themselves, as well as high mechanical loads. Further, high heat flux and electromagnetic fields from the plasma impose their own electrical and thermal loads on magnetic components. These combined stresses impact the safety considerations and designs of a given magnet (and its associated supporting systems) for a given MCF device configuration. To increase efficiency, improve safety, and reduce experimental cost during design, it is important to accurately model these magnet systems at device scale. In 2024, research efforts were begun to do just that: develop large-scale, integrated, multiphysics magnet simulation capabilities within the Multiphysics Object Oriented Simulation Environment (MOOSE) framework, developed at Idaho National Laboratory (INL).

In this talk, we will present progress made in developing this simulation capability, which is based on a combination of several established MOOSE physics modules: electromagnetics, heat transfer, solid mechanics, and thermal hydraulics. Current model development efforts are focused on improving the MOOSE electromagnetics module to enable studies using vector and scalar potentials, as well as calculation of mechanical stresses and deformation in the magnet during operation. The multiphysics coupling approach as well as current benchmarking, verification, and validation exercises will be outlined. Looking ahead, we will conclude with a discussion of planned model development efforts focused on MOOSE-based simulations of magnet cooling and power delivery systems.

Commissioning and first measurements of the visible light diagnostics of the SMART tokamak

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Optical diagnostics are among the main diagnostics foreseen for the first operational phase of the SMall Aspect Ratio Tokamak (SMART) [1]. SMART is a spherical tokamak (ST) operating at the University of Seville that has recently achieved its first tokamak plasmas. The coils system of SMART has been designed to allow flexible shaping of the plasma column, being able to accommodate plasmas in both positive (PT) and negative triangularities (NT). Fast cameras facilitate high temporal resolution images of the discharge while spectrometer-based optical diagnostics, such as Charge eXchange Recombination Spectroscopy (CXRS) [2], provide measurements of the plasma species properties.

This contribution presents the first measurements of the recently installed gas puff based CXRS diagnostic [3], that allows measuring the density, temperature and toroidal rotation velocity of different plasma species. A fast controllable piezo valve puffs a neutral gas (for example, but not limited to, helium or hydrogen) into the plasma. Then, the neutral gas emits light after CX reactions with the main or impurity ions of the plasma. An optical system captures the emitted photons from 25 lines of sight into different optical fibers that feed the signal into a spectrometer with a resolution of $\Delta\lambda \approx 0.0167$ nm/pix@ $\lambda = 470$ nm.

Additionally, in this contribution, the first measurements of SMART discharges using a Phantom® v2512 high speed monitoring camera are presented. This camera is capable of capturing images with a temporal resolution of up to 1 MHz. Using a wide-angle lens, most of the plasma region is covered, as shown in the example below. This visible fast camera allows for monitoring the discharge, the plasma facing components and inferring the plasma position.

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Influence of bridge installation layout on eddy current loss and optimization strategy in strong electromagnetic environment

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Wednesday Posters 1, Lobdell (Building W20 Room 208), June 25, 2025, 10:30 AM - 12:00 PM

With the increasing demand for commercialization of the nuclear fusion industry, the design, selection and installation of bridges in fusion-related facilities must meet the standards of the fusion industry. There are a large number of high-current and high-magnetic-field equipment in fusion facilities. The bridges around the equipment often generate local high temperatures due to eddy currents and hysteresis effects, threatening the safe operation and service life of the system. To address the above problems, this paper conducts an in-depth study on the influence of electromagnetic fields on bridge installation design. Through theoretical analysis, numerical simulation and experimental verification, the temperature rise characteristics of bridges under different materials, shapes and layouts are systematically explored. The research results show that:
Reasonable selection of bridge materials and structures, such as using materials with lower magnetic permeability or optimizing the cross-sectional shape of the bridge, can effectively reduce eddy current losses;
In terms of installation layout, increasing the distance between high-current cables and bridges, and adopting layered dislocation or shielding measures can reduce the induced current and local temperature on the bridge surface;
Optimizing the grounding method and overlap position of the bridge can help to reasonably divert the eddy current path and further suppress local overheating.
Based on the above research, this paper proposes a bridge selection and installation scheme that takes into account both safety and economy, which can provide a reference for reducing high temperatures caused by eddy currents and hysteresis in actual projects. This research is of great significance to improving the reliability of fusion equipment and extending its service life. It has made certain explorations in the standardization of installation design and selection, and has also laid the foundation for further in-depth discussion of electromagnetic compatibility issues under high current environments.

Lithium Wetting Behavior on Pressed and Sintered Stainless Steel-316 Felt Material for Enhanced Evaporator Performance in LTX-β

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Wednesday Posters 1, Lobdell (Building W20 Room 208), June 25, 2025, 10:30 AM - 12:00 PM

The interaction of liquid lithium with plasma-facing components (PFCs) is essential for improving fusion energy systems by enhancing plasma performance and reducing wall-material erosion [1]. This study investigates the wetting behavior of lithium on pressed and sintered stainless steel-316 felt-like material, a candidate material for the upgraded evaporator system in the Lithium Tokamak Experiment-β (LTX-β) at Princeton Plasma Physics Laboratory (PPPL). It aims to determine the minimum wetting temperatures of untreated and treated samples, with the latter exposed to ultrahigh vacuum (UHV) at 1100 °C, and further evaluates the role of hydrogen partial pressure in reducing surface oxides and improving wetting behavior. Previous studies [2] have demonstrated that the structural properties of stainless steel enhance capillary action, enabling uniform lithium distribution, and that hydrogen treatments can reduce wetting temperatures below 315 °C. This work employs the liquid metal dropper system [3] developed at The Pennsylvania State University to facilitate precise lithium droplet deposition and support advancements in plasma-facing component technology.

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Design and Development of an Optical Gas Sensor for Fusion Applications

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Wednesday Parallel 2a - Diagnostics, Instrumentation, Data Acquisition & Management II, Kresge Main Theater (Building W16, upstairs), June 25, 2025, 2:00 PM - 3:30 PM

Fast and accurate neutral gas measurement systems will be critical for the realisation of future deuterium-tritium (D-T) fusion reactors. This is required for the closed loop fuelling cycle of the reactor, where quantities of exhaust fuel gases, consisting primarily of isotopes of hydrogen (H) and helium (He), are monitored in real time.

Typically, quadrupole mass spectrometry (QMS) is employed to measure gas partial pressures, however the very similar mass-to-charge ratios of fusion gas species makes this measurement using QMS extremely challenging. For example, the masses of D_2 and 4He are separated by 0.02 amu. Techniques such as threshold ionisation mass spectrometry can be utilised to separate closely spaced masses, however this method has difficulty resolving low concentrations with adequate speed.

An alternative route, using remote optical emission spectroscopy (ROES), was demonstrated by Klepper et al[1] and this is seen as a promising method for overcoming the inherent mass measurement problem encountered by QMS. ROES involves the generation of a small, remote plasma which is used to excite gaseous species into emitting light, which can then be measured by an optical spectrometer and the gases identified and quantified by their light emission.

Whilst ROES is an extremely promising technique it is not without its challenges for use in fusion applications. Whilst D_2 and ^4He light emissions are separated by > 10 nm, the isotopic emissions of He and H are very closely spaced, requiring high resolution optical spectroscopy. Furthermore, the sensor will be required to operate in reactor fringing fields of more than 0.2T whilst maintaining a stable plasma within the sensor. A further complication is the inherent presence of ionising radiation and neutrons produced by the fusion reactor.

In this paper we present the development and design of a ROES sensor for fusion applications. The sensor is qualified in its ability to detect small (<0.1%) concentrations of H and He gas isotopes (D/T and ^3He/^4He) with a speed of response of less than 1 second. This surpasses the requirement for the future fusion reactor, ITER. Optimisation of the sensor's plasma for resolving closely spaced emissions will be presented. Finally, the stability of the sensor's operation in a representative fusion environment is discussed. Experimental results of sensor operation during exposure to magnetic fringing fields, gamma radiation and neutron bombardment (up to 0.3 T, 500 kGy and 1×10^{14} n/cm^2 respectively) will be shown.

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AI for Fusion: Possibilities and Applications in Enhanced Diagnostics, Control and Science Discovery

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Wednesday Parallel 2a - Diagnostics, Instrumentation, Data Acquisition & Management II, Kresge Main Theater (Building W16, upstairs), June 25, 2025, 2:00 PM - 3:30 PM

Advancements in AI are driving transformative progress in the field of fusion engineering. I will provide an overview and highlight the recent achievements of our group in this area: 1) Robust plasma state prediction even when there is diagnostic failure 2) Finding the minimal set of diagnostics needed to operate a reactor 3) Fusing data from multiple diagnostics (new Multimodal Super-Resolution technique) to discover hidden physics insights 4) Prediction of plasma evolution in future fusion reactors such as ITER by combining experimental data and simulations from multiple machines (DIII-D, AUG; ASTRA, TRANSP, TGLF,...) 5) Reinforcement learning control that achieves high performance fusion reactor operation without instabilities (tearing modes and ELMs) both at KSTAR and DIII-D.

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Design and construction of a new Fast Reciprocating Manipulator (FaRM) for RFX-mod2

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Wednesday Parallel 2a - Diagnostics, Instrumentation, Data Acquisition & Management II, Kresge Main Theater (Building W16, upstairs), June 25, 2025, 2:00 PM - 3:30 PM

A substantial modification of the toroidal complex of the RFX experiment, named RFX-mod2, is currently under completion, involving the whole core system of the machine and in particular the vacuum vessel, the entire plasma facing components and a wide set of in-vessel diagnostic systems.
In the experimental stages of the RFX-mod2 experiment, the measurement of plasma parameters, such as density, temperature, plasma potential, and magnetic field in the edge region, will be of paramount importance.
In cases where the plasma current is higher than 0.5 MA, it is necessary that the insertion of the measurement probes is carried out in a very short period, to avoid damage to the probes due to high thermal fluxes. This can be accomplished by means of a fast reciprocating manipulator, which is capable of inserting, holding in place and removing the diagnostic head in very short time.
The Fast Reciprocating Manipulator (FaRM) for RFX-mod2 will allow the insertion of a diagnostic head weighing about 2 kg for times on the order of tenth of a second inside the RFX-mod2 chamber, working in the presence of strong and rapidly changing magnetic fields. In particular, an insertion time of less than 50 ms and a removal time of less than 50 ms are required. The dwell time in the stretched position will be settable between 50 and 200 ms.
This work summarizes the design and construction of FaRM.

Design of the Diagnostic Residual Gas Analyzer system for the Material Plasma Exposure eXperiment

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Wednesday Parallel 2a - Diagnostics, Instrumentation, Data Acquisition & Management II, Kresge Main Theater (Building W16, upstairs), June 25, 2025, 2:00 PM - 3:30 PM

The Material Plasma Exposure eXperiment (MPEX) is a linear plasma device under construction at Oak Ridge National Lab. The mission of MPEX is to study the evolution of material surfaces, including pre-irradiated materials, in steady-state fusion reactor-relevant diverter conditions with ion flux up to ~10^24 m^-2 s^-1, ion fluence up to ~10^31 m^-2, heat flux up to ~40 MW m^-2, and neutral pressures up to 10 Pa. The diagnostic set of fusion reactors, such as ITER, include Diagnostic Residual Gas Analyzers (DRGAs), to measure the evolution during plasma discharges of neutral gasses in the main chamber and divertors. The initial diagnostic set for MPEX includes a DRGA system, based on the ITER DRGA design, capable of steady-state, real-time monitoring of the evolution of fusion relevant neutral gas (H2, D2, He, Ne, etc.) partial pressures at the MPEX target. The Final Design Review (FDR) of MPEX diagnostic systems was recently completed, including the MPEX DRGA. The DRGA instrumentation platform is modular and will be located outside the 50 Gauss contour to minimize the magnetic shielding required. The gas conductance timescales from the MPEX device to the DRGA diagnostic were calculated to be <~1 s, consistent with MPEX and ITER divertor DRGA design requirements. The design, calculations, lab-testing, and installation status of the MPEX DRGA components will be given in this presentation.

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The German ReFus project – regulation of fusion facilities

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Wednesday Parallel 2b - Safety, Regulation, and Neutronics II, Sala de Puerto Rico (Building W20 Room 202), June 25, 2025, 2:00 PM - 3:30 PM

The ReFus project on the regulation of fusion facilities in Germany has been launched by the German Federal Ministry of Education and Research. It aims to analyze the status quo and possible gaps in the existing German legal framework and the technical regulatory requirements for fusion facilities and future fusion power plants. Based on this analysis it will be identified if some of the existing legal framework can be adopted for fusion, and go a step further, if a special fusion legal framework is mandatory, and by needs, how to develop a concept for the technical regulation of fusion facilities. The project is contributed by German TSOs and research organizations and supported additionally by eight startup/industrial companies as associated partners. In this way, a consensus for fusion regulation should be strived for by all parties if possible. If different opinions remain, the differences will be documented.
We are in the first phase of the project focusing on a survey of the needs of the startup and industrial stakeholders. Together with an analysis of the existing legal framework for radiation protection and nuclear facilities in Germany, the project evaluates how fusion facilities can either be included explicitly in current legal frameworks, especially the one for radiation protection regulation, or if a special legal framework for fusion facilities should be developed, and how such a legal framework for fusion regulation should look like. This will include the topics radioactive waste and non-proliferation.
We have also started the second phase during which we evaluate the existing national and international practices for anticipated fusion regulation, experiences in licensing, operating and decommissioning of existing fusion facilities to development a concept for the technical regulatory requirements. Moreover, the identified hazard potential of fusion facilities is compared with the licensed facilities under existing regulations such as highly radioactive sources in medical or industrial applications under radiation protection regulation or inventories of nuclear facilities under nuclear regulation. Based on this evaluation of hazard potential the technical requirements for safety measures can be derived and justified.
For safety demonstrations of fusion power plant, requirements for safety computer codes are inevitable. They will very likely be included in the technical regulatory requirements. Therefore, in a third phase, a gap analysis is performed if the existing codes satisfy the needs of safety analyses in fusion application and requirements are identify for codes extensions or developments according to priority.
The project is sponsored by the German Federal Ministry of Education and Research in the frame of the program FUSION 2040 – Research on the way to a fusion power plant (contract number 13F1002A).

OpenMC, a one-stop-shop for fusion neutronics: 2 years later – a use case for fusion power plant nuclear design and analysis

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Wednesday Parallel 2b - Safety, Regulation, and Neutronics II, Sala de Puerto Rico (Building W20 Room 202), June 25, 2025, 2:00 PM - 3:30 PM

Myriad nuclear analyses are required in the fusion power plant design process. These analyses determine critical system specifications and requirements yet are extremely challenging to execute while providing rigorous uncertainty quantification to establish engineering margins. Two years ago, at SOFE 2023, we presented on the roadmap for a one-stop-shop for fusion neutronics, based on the open-source Monte Carlo code, OpenMC, and the Rigorous 2-Step (R2S) methodology. Here we present the progress towards this goal, demonstrating the new and freely-available capabilities for fusion power plant design including prompt and delayed responses due to neutron transport in a compact tokamak power plant. Demonstrated analyses include prompt nuclear heating from neutrons and photons, site dose and shielding calculations with automated variance reduction techniques, activation and waste classification, as well as shutdown dose rates. All analyses are performed on CAD-based models using DAGMC transport algorithms and unstructured mesh tallies. Performance comparisons are presented for select analyses between contiguous and discontiguous meshes and ongoing development of capabilities for fusion power plant uncertainty quantification are discussed.

Impact of operating characteristics on impurity specifications for low-activation fusion materials

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Wednesday Parallel 2b - Safety, Regulation, and Neutronics II, Sala de Puerto Rico (Building W20 Room 202), June 25, 2025, 2:00 PM - 3:30 PM

While fusion has the potential to avoid production of long lived radioactive waste, it will nevertheless produce a large amount of radioactive material via neutron activation of surrounding structures and materials, which has the potential to detract from its attractiveness as an energy source. A key objective for fusion is to ensure that components do not exceed criteria for low level waste (LLW), which is defined by Nuclear Regulatory Commission (NRC) in 10 CFR 61. This regulation addresses only a small subset of radionuclides that might result from neutron activation; an extensive effort has been conducted in the past to extend the list to include all those relevant to fusion applications. To satisfy the LLW requirements, all fusion materials should be carefully chosen to minimize alloying with Al, N, Ni, C, Cu, Nb, Mo, Re, Ag, etc., which generate long-lived radionuclides such as C-14, Ni-59, Nb-94, Mo-99, etc. Equally important, specific impurities (such as Nb, Mo, Ag, Re, etc.) must be controlled to a low level to avoid generating waste exceeding LLW criteria (so called “greater than class C” waste).

In this work, we define impurity specifications for fusion materials that future manufacturers must meet in order to satisfy LLW criteria. Because the activity density depends on the device size, power, and operating time, we conduct parametric activation analyses using a tokamak blanket model to build the specification as a function of these parameters. In addition to isotopes affecting long-term waste disposal, we consider additionally impurities with intermediate half lives, and their implications for potential decay storage and recycling schemes that would greatly reduce the volume of waste produced from these materials, the prospects of which are improved if low impurity levels can be achieved.

An Open-Source, CAD-Based Validation Platform for Fusion Neutronics Benchmarks

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Wednesday Parallel 2b - Safety, Regulation, and Neutronics II, Sala de Puerto Rico (Building W20 Room 202), June 25, 2025, 2:00 PM - 3:30 PM

Neutronics analysis remains a critical bottleneck in fusion reactor design cycles due to challenges converting complex computer-aided design (CAD) models to viable geometry representations for neutron transport simulations. As the complexity of fusion reactor designs grows, alongside increasingly intricate engineering workflows, the integration of CAD-based geometries into neutronics workflows has become essential. However, these integrations are fraught with challenges and require rigorous testing and robust validation frameworks to ensure reliability.

This presentation introduces the first comprehensive benchmarking repository featuring high-quality CAD models of fusion-relevant experimental benchmarks. These CAD models are derived from automatic conversions of experimental data available in prominent V&V databases, such as SINBAD and CONDERC. The repository supports advanced workflows, including the Total Monte Carlo (TMC) method for uncertainty quantification and facilitates the comparison of transport solutions on constructive solid geometry (CSG), direct accelerated geometry Monte Carlo (DAGMC), and unstructured mesh (UM) geometries derived from the same CAD model.

Leveraging OpenMC's open-source framework, this repository provides an automated validation platform for fusion neutronics analyses starting from CAD models. The repository not only offers open-source validation tools but also provides detailed quality assessment reports for benchmarks, workflows, and neutronics codes. By addressing the challenges of CAD integration, this work provides a unified platform that strengthens the capabilities of fusion reactor neutronics, pushing forward the standards for validation and benchmarking in the field.

Fusion Structural Materials – Progress, Challenges and Design Impact

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Wednesday Parallel 2c - Materials and Materials Systems; Codes/Standards for Fusion, Kresge Little Theater (Building W16, downstairs), June 25, 2025, 2:00 PM - 3:30 PM

The development of structural materials for fusion energy represents one of the most critical challenges on the path to delivering viable fusion power. Structural materials must withstand the extreme environment of a fusion reactor, which combines intense neutron irradiation, high temperatures, high heat flux, plasma particle flux and chemically reactive coolants, while maintaining mechanical integrity and minimizing activation and waste disposal challenges. This talk will present the key advances in fusion structural materials, focusing on Reduced Activation Ferritic-Martensitic (RAFM) steels, Oxide Dispersion Strengthened (ODS) steels, and Vanadium alloys, with emphasis on their performance under irradiation, their evolving design criteria,. RAFM steels based on 9Cr-WVTa system have long been considered the baseline structural materials for near-term devices with water-cooling, and/or PbLi blanket systems due to their relatively low activation, good mechanical properties, and established fabrication routes. However, limitations such as low temperature hardening-embrittlement (LTHE) and reduced high-temperature strength still remain a major challenge. Moreover, a lack of understanding of the effect of He on irradiation embrittlement remains an issue. Here, results from neutron irradiation over the last decade will be compiled to revisit our understanding of irradiation embrittlement in RAFM steels, including the effect of Helium. The results will be compared with neutron irradiation data on 12-20% Cr ODS alloys to map LTHE effects between RAFM & ODS alloys. The results will also showcase new aspects of nanoprecipitate instability under neutron irradiations in this class of materials. Lastly, for liquid Li blanket systems, the properties of Vanadium alloys (V-Cr-Ti) will be compared against RAFM/ODS steels to highlight the pros and cons of materials choice, and its effect on design.

Materials and Manufacturing After Ignition: How Target Fabrication Is Enabling Inertial Confinement Fusion

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Wednesday Parallel 2c - Materials and Materials Systems; Codes/Standards for Fusion, Kresge Little Theater (Building W16, downstairs), June 25, 2025, 2:00 PM - 3:30 PM

The demonstration of inertial fusion ignition on NIF in 2022 was a triumph of integrated physics at a grand experimental scale. Underlying this physics demonstration was the unique and complex mix of materials science and manufacturing technology needed to produce an igniting inertial confinement fusion (ICF) target. While the basic chassis of the ICF target has remained unchanged since the National Ignition Campaign of 2010, material improvement to the hollow capsule that contains the fusion fuel was a key ingredient to ignition. Years of ICF experiments tell us that practical performance limitation of capsules is in part defect-driven. Ignition and its follow-on experiments provided important empirical evidence on which defects affect performance in the burning plasma regime. LLNL, General Atomics, and Diamond Materials GMBH have used this knowledge to accelerate the rate of learning on NIF in two ways.

First, we developed production, metrology, and data management workflows to increase the availability of high-quality capsules for NIF experiments – without needing to improve the underlying capsule quality! This was accomplished by adopting a high-throughput front end and high-yield back-end production model, with matched fabrication and metrology capacities. New metrology capabilities were needed to connect the front- and back-ends as well as process improvement for a reliably high-yield back-end. Second, we focused material science developments on reducing the density of defects observed to drive implosion instabilities (e.g., bulk voids, surface pits, and non-concentricity). Reducing these defects required detailed studies of coating processes on non-stationary spherical substrates, polishing round surfaces, and optical and x-ray metrology. Notably, machine learning techniques were a key tool used in these developments. Advances in these areas have increased the fraction of capsules meeting ignition requirements from <10% in 2023 to an expected >30% in 2025.

These improvements in materials science and manufacturing workflow have helped NIF experiments increase implosion yields from ~1 MJ in 2021 to >5 MJ in 2024 and position ICF to explore higher yields that will be made possible by future laser upgrades to NIF. These results highlight the importance of experimental validation of requirements in accelerating progress in regimes where defects limit performance. This experience also provides a perspective on how ICF target fabrication – which is focused on supporting design diversity and scientific understanding of individual experiments – can inform future inertial fusion energy (IFE) targets which will need statistical process control, volume scaling, and unit cost.

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Advancements in pressure codes & standards for fusion power plants

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Wednesday Parallel 2c - Materials and Materials Systems; Codes/Standards for Fusion, Kresge Little Theater (Building W16, downstairs), June 25, 2025, 2:00 PM - 3:30 PM

The purpose of design codes and standards is to establish national or international standards that consist of a set of rules based on state-of-the-art knowledge, experience, and experimental feedback from facilities. The design and construction of any fusion reactor should make use of appropriate codes and standards to provide quality assurance and control for the structural integrity and safety of these plants, such as pressure vessels which cover vacuum vessels, breeder blankets, and high-pressure cooling components. The codes provide the bridge between different suppliers, participants, researchers, designers, manufacturers, and regulators. The documents can be viewed as a live document that are updated as better operational experience, knowledge, and scientific advancements become available.

The first edition of American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel (BPV) Section III Division 4 "Fusion Energy Devices" construction code and standard was released in June 2023. In 2024, Division 4 released its 5-year strategy to develop a construction code and standard for pressure vessels for fusion power plants. This marks the beginning of the activities, and one task is to write the materials qualification requirements to bring in "fusion-grade" structural materials within the Division for use in the future.

This presentation has three objectives: 1) provide update on the development of the codes and standards for fusion power plants 2) outline a proposal for the materials qualification route for Division 4 which outlines the testing requirements, standards, assessment methodologies, environmental effects such as corrosion and irradiation, and 2) receive feedback from the fusion community on the approach. It is important that the code reflects the best practice and community needs/requests, and this talk is the first of many outreach activities Division 4 will be conducting over the years to ensure the construction code is fit for purpose for if/when fusion power plants become a reality.

Power-sharing simulations of near-double null plasmas in ST40

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Wednesday Posters 2, Lobdell (Building W20 Room 208), June 25, 2025, 2:00 PM - 3:30 PM

We present a new modeling methodology for the simultaneous prediction of 3D surface temperature evolution at all four divertor targets in near-double null tokamak configurations, which is especially important for compact high-field devices that may not have the ability to dissipate large amounts of power on the high field side. The private fusion energy company Tokamak Energy Ltd., based in Oxfordshire in the UK, owns and operates the high-field spherical tokamak ST40 ($B\phi \leq 2.1$ T, $A \geq 1.6$). It is equipped with a set of up-down symmetric (but highly non-axisymmetric) molybdenum divertors, typically operating in a disconnected double-null (DDN) configuration [1]. Evaluating the power-sharing between the four divertor strike points in the DDN configuration is important for understanding the overall power balance in ST40, as well as for optimizing the power exhaust performance and prolonging the survivability of the plasma facing components. This power-sharing is typically evaluated in terms of the separation between the primary and secondary separatrices at outer midplane, dRsep. The free-boundary Grad-Shafranov solver FreeGS [2] is used to generate plasma geometries with $-4 \leq$ dRsep ≤ 4 mm, within the typical range of dRsep control in ST40. The power sharing between the divertors is then prescribed using a variation of Brunner's power fraction model [3]. These calculated power fractions are then used as inputs to the Heat Flux Engineering and Analysis Toolkit (HEAT) [4], which calculates the deposited heat flux on the 3D divertor targets and their resultant temperature change. Initial results have shown that a significant amount of power, ~12 - 50%, goes to the inner divertors for lower biased DDN configurations with Eich type heat flux profiles, which may imply detrimental effects if sustained for a long period of time without additional dRsep control. The simulation results will be compared with available experimental data from divertor infrared thermography, Langmuir probes and radiated power diagnostics.
Work supported by US DOE under contract DE-AC05-00OR22725.

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Prospects of ‘Liquid Stellarator Reactors’ for enhanced competitiveness

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Wednesday Posters 2, Lobdell (Building W20 Room 208), June 25, 2025, 2:00 PM - 3:30 PM

Full-liquid in-vessel stellarator reactors have the potential to significantly reduce the construction and operational cost of fusion power plants and improve their competitiveness.
The work integrates magnetic configurations, centrifuge-breeding liquid walls, liquid lithium surfaces, and a novel divertor concept (the Distributed Divertor) in stellarator reactors. The potential and challenges for these stellarator reactors are reported.
The Distributed Divertor is a type of non-resonant divertor, in which the power is uniformly distributed on a specially-shaped toroidal surface called ‘Equi-power surface’, instead of concentrating the power. It leverages the pumping capability of lithium and the particular structure of stellarator magnetic fields outside the Last Closed Flux Surface.
In conventional reactor designs, structural blankets and divertor targets need to be replaced every few years, thus, requiring new blankets/divertors, large remote maintenance equipment and massive hot cells for subsequent storage (more replacements for high-field high power-density reactors). Certainly, the consequent cost may hinder their competitiveness against other energy sources.
To address these challenges, this work investigates and compares two approaches of stellarator reactor. One approach combines centrifuge molten salts located at the straight stellarator sectors, a high-mirror QI magnetic configuration (to partially concentrate neutrons), and liquid-lithium divertor targets located at the curved sectors (V. Queral et al., IEEE Trans. Plasma Sci. 52, 2024). The other combines centrifuge molten salts covering the full toroid, a softly-shaped magnetic configuration and a liquid-lithium Equi-power surface (a Distributed Divertor) located at the straight sectors.
To assess such concepts, experiments performed with centrifugal water, centrifugal galinstan and lithium floating on molten salt are reported. In addition, the calculation method for a Distributed Divertor and an example of it are presented.
The advantages, challenges and Technology Readiness Level of both reactor approaches are outlined.

Bridging the Fusion gap with FAST: Pioneering Power Generation and Tritium Breeding

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This paper presents the outline of the FAST (Fusion by Advanced Superconducting Tokamak) project and its initial design. The major objective of this plant is to demonstrate fusion power generation in 2030's and to provide a platform to develop, evaluate and improve the maturity of fusion energy technology that will be essential for power plants. Since integrated fusion nuclear technology has not been developed under relevant conditions to date, to fill the gap is required to reduce the technical risk. FAST is therefore intended to provide the reactor relevant volumetric neutron flux with facing burning DT plasma for technically meaning time scale. Advanced low aspect ratio tokamak configuration with high temperature superconducting magnets is designed. Typical target performance is, fusion power by DT fusion reaction: 50-100 MW, neutron wall loading: 0.3-1 MW/m2, discharge time : 1000 sec, and total full-power operation life time of about 1000 hours. Breeding blanket with coolant will be equipped with fully functional thermal energy cycle and tritium extraction process to verify their function under relevant condition. Power conversion system will generate electricity as well as other energy applications such as hydrogen production from high grade heat. Tritium breeding ratio from fusion neutron to produce tritium fuel will be measured and demonstrated in 2030s as expected by the Japanese Government. However, it should be noted total net energy gain and tritium self-sufficiency is measured but not committed.

Quasi zero-dimensional parameter survey identified a possible design window to satisfy above requirements with a low aspect ratio tokamak of major radius 2-3m with external heating and current drive of ca.50MW. Hybrid scaling of interpolation from high and low aspect ratio scaling suggested relatively modest plasma parameters can achieve require performance, and design optimization is carried out to minimize the cost and construction period within acceptable technical risks, that could be affordable with private sources with public supports. Integrated fusion fuel cycle and safety features as an energy plant that will fill the technical gap toward energy generation plants. Because of the thin shields in the in-vessel components, machine life time and some of the performance are compromised.

Plasma and engineering of the machine are designed by the collaboration by private industry, universities and research institute professors and researchers domestically and internationally. Major companies in Japan from various fields such as construction, trading, supply chain and international partners support the project. The conceptual design is performed within 2025 followed by site selection process. Evaluation of the project on the technical feasibility, funding, regulation and policy, will be conducted at the transition to detailed design, where a decision will be made for execution. The facility is expected to be commissioned in 2035.

Research and Development on Future Megawatt-class Gyrotrons in Europe

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In Europe significant research and development is underway to build and further advance megawatt-class gyrotrons for current nuclear fusion experiments and the future European DEMOnstration power plant (EU DEMO). While the gyrotron R&D is being carried out by members of the European GYrotron Consortium (EGYC), namely the Karlsruhe Institute of Technology (KIT), Germany, the Swiss Plasma Center (SPC) at EPFL, Switzerland, the Kapodistrian University of Athens (NKUA), Greece, and the National Research Council in Italy (ISTP-CNR), the validation of industrial gyrotrons in a fusion relevant environment is strongly supported by IPP Greifswald, Germany that operates the stellarator Wendelstein 7-X. This is complemented by CEA, France, which is currently installing its new ECRH system for the WEST tokamak. Finally, the gyrotron R&D is supported by Polytechnic of Turin (PoliTO) and the Polytechnic of Milan (PoliMI). European industrial partner THALES (France) is responsible for industrial design and manufacture.
The first generation of megawatt-class industrial gyrotrons developed in Europe is based on the 140 GHz, 1 MW gyrotrons of W7-X (THALES 1507). The same applies to the 170 GHz, 1 MW gyrotrons of ITER and DTT; the 126/84 GHz, 1 MW dual-frequency gyrotron from TCV, Switzerland; the 105 GHz, 1 MW gyrotron from WEST, France; and, the 117.5 GHz, 1 MW gyrotron of DIII-D, USA. Building on this first generation of gyrotrons, KIT, NKUA and IPP Greifswald, are moving on to the next generation of W7-X gyrotrons that will deliver an output power of 1.5 MW. The aim is to move up to a 2 MW class gyrotron before 2030. While the fundamental designs are still based on hollow-cavity technology, within EUROfusion, the European R&D is focusing on the 2 MW coaxial-cavity gyrotron technology. This technology will make it possible to obtain multi-purpose, multi-frequency gyrotrons that will operate at various possible operating frequencies ranging from 136 GHz to 238 GHz. Combined with an ExB drift Multistage Depressed Collector (MDC), it should enable gyrotron efficiencies above 60 %. In frame of an EUROfusion Enabling Research (ENR), KIT and NKUA are working on possible concepts for efficient, megawatt-class gyrotrons that are capable to operate at the 2nd harmonic of the electron cyclotron frequency. All of those developments are supported by ongoing advances in the simulation tools and validation capabilities, e.g. the new FULGOR gyrotron teststand at KIT.
This presentation gives an overview of advances in gyrotron research and development in Europe.

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Robust model predictive density profile control with discrete pellets, applied to integrated simulations of ITER

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Wednesday Posters 2, Lobdell (Building W20 Room 208), June 25, 2025, 2:00 PM - 3:30 PM

Achieving the peaked core electron profiles necessary for high-performance scenarios in future fusion tokamaks such as ITER requires safe density control with pellet fueling, which means never violating scenario-specific density limits such as the edge-localized Greenwald density limit. ITER will primarily rely on injection of large frozen pellets containing deuterium and tritium to fuel the plasma core. Due to the size of ITER fuel pellets, they must be treated as discrete actions for safe, high-performance density regulation, complicating the density profile control problem significantly. To design and test safe density profile control with pellet fueling, we couple the controllers with JINTRAC integrated simulations of the ITER 15 MA/5.3 T scenario. In doing so, we must grapple with the reality that turbulent transport dynamics are not well understood and difficult to model. Indeed, the turbulent transport models available for practical integrated simulation of core plasmas with discrete pellets yield widely varying solutions for the evolution of the density profile, given the same initial conditions. As we cannot determine which transport model (e.g. TGLF or Bohm/gyro-Bohm) is closer to the ground truth of ITER plasma turbulent transport, truly safe controller design should be able to achieve density profile control regardless of the transport solver utilized. We propose a predictive density profile controller that considers fuel pellets as discrete actuators, ensures that safety-critical density limits are not violated, and is robust against unmodeled transport dynamics. The model predictive control (MPC) scheme we deploy combines the physics informed dynamic mode decomposition approach to learn a prediction model from high-fidelity simulation data, our novel modified penalty term homotopy algorithm for real-time MPC (PTH-MPC) with discrete pellets, and compares two robust MPC approaches (multistage scenario MPC and tube-based robust MPC) to ensure unmodeled pellet ablation and transport dynamics does not result in safety limit violation. We demonstrate safe density profile control with discrete pellets by coupling the robust PTH-MPC density controller with two sets of JINTRAC integrated simulations of the ITER 15 MA/5.3 T scenario. The first uses the Bohm/gyro-Bohm semi-empirical turbulence model, while the second utilizes the more first principles physics-based TGLF-SAT3 gyro-Landau-fluid model. Both simulations use HPI2 to model discrete pellet deposition and ablation. We demonstrate that we can achieve safe density control with robustness guarantees in real-time while treating pellets as discrete events. We discuss the benefits and limitations of robust PTH-MPC in density profile control, and propose possible improvements for future controller design.

Evaluation of plasma behaviours using low Z materials for wall conditioning in EAST

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Low-Z materials are widely employed for wall conditioning to enhance plasma performance in fusion devices. In EAST machine, lithium (Li) evaporative coatings significantly reduced both the impurity concentration and fuel recycling in the plasmas [1]. Specifically, the high-Z tungsten core impurity concentration was maintained between 3 ppm–15 ppm, H/(H+D) <5%, and global recycling coefficient<1 during >100 s H-mode discharge. However, it also demonstrated routine Li coating has short lifetime about 300s. Therefore, real-time wall conditioning via feedback Li powder injection was used to further control fuel recycling and reduce the core high-Z metal impurities during >600s long pulse H-mode discharge in 2024. In addition, for supporting ITER new baseline, a series of boronizations were also carried out in EAST. It was found that the impurity radiation including oxygen and heavy impurities such as W, Fe, Cu and Zeff decreased significantly, which results in the slightly increased plasma stored energy. The lifetime of boronization was about 1700s [2]. However, the H release was very serious during the initial plasma discharges after boronization due to H co-deposition during boronizaiton [3]. To avoid introducing H isotopes, pure boron (B) powder was injected into plasma for real time B coating[4], and the W impurity content could be decreased to 10-5 as B powder continuously injecting. By analyzing the fuel particle retention, it was found the and each B atom during B powder injection exhibited a trapping capacity 0.3 D [5] particles, which was lower than that during Li powder injection with the ratios of > 0.4.
Compared Li coating, B coating has similar impurity suppression capability and longer lifetime, but plasma confinement has a degradation of >15%, possibly due to higher hydrogen release after B coating by using carborane (C2B10H12). These advances provide a very valuable reference for evaluating low Z materials application in ITER and future fusion reactor devices.

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High Current Tokamak Protection Switch

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Wednesday Posters 2, Lobdell (Building W20 Room 208), June 25, 2025, 2:00 PM - 3:30 PM

The tokamak, a leading architecture for future fusion power systems, faces many challenges prior to successful operation. One of the major technical hurdles is plasma disruption events, which can result in the generation of high current beams of relativistic electrons (10s of MeV). These beams are called Runaway Electrons (RE). Runaway Electrons can cause severe damage to plasma-facing surfaces of the tokamak structure, such as melt damage, coolant leaks, and loss of vacuum. Even if RE events occur rarely, they could prevent fusion machines from reaching commercial viability. High field tokamaks, required for commercial fusion power generation, will be even more susceptible to damage from RE than experimental systems, such as ITER.

To prevent these events from damaging tokamaks, a non-axisymmetric coil can be excited to disrupt the magnetic field, and prohibit formation of RE beams. Diversified Technologies, Inc. (DTI) is working under a Small Business Innovative Research (SBIR) grant from the Department of Energy (DOE) to develop a fast-acting high current switch and vacuum feedthrough controlling a non-axisymmetric coil. When passively switched ON, this will disrupt formation of the relativistic beams, and prevent damage to the plasma facing surfaces. Protective circuits are integrated into the switch to ensure proper operation during RE events.

The full-scale switch and feedthrough will be installed in a working fusion device for full-scale tests. This effort is in collaboration with MIT for future installation on prototype devices such as DIII-D. This paper will discuss the high current switch design, latest full-scale 350 kA, 5 kV experimental switch progress and data, and plans for future testing and installation at DIII-D.

Flyer Plate Valve Power Supply for the ITER Shattered Pellet Injection System

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The design of the flyer plate valve power supply for the ITER shattered-pellet-injection (SPI) disruption-mitigation system (DMS) has been revised, fabricated and tested. This new design incorporates system improvements towards higher reliability, improved testability, and reduced size and weight. The primary electronic-drive module, a high-voltage pulsed power supply (HVPPS), features a programmable, high-voltage dc power supply (0-3kV), an energy-storage capacitor bank (4 x 100?F), and a pair of SCR-based high-current switches with separate isolated trigger circuits. Proper selection of the dc voltage and sizing of the capacitor bank ensures timely and complete opening of the flyer plate valve (FPV) using a wide range of cable lengths.

The updated design incorporates several features to promote increased reliability, testability, and observability. Redundant trigger and firing circuits are incorporated for the critical firing path. High current paths have been replaced with copper bus bars providing improved electrical and mechanical connectivity and durability. Integrated sensors facilitate drive optimization enabling a single hardware design to be adapted for use with a wide range of cable lengths. The unit is configured to drive an external FPV load, or an internal FPV coil, enabling in-situ health assessments over the system lifetime. Measured parameters include dc resistance monitoring of the flyer plate coil, output relay, and associated transmission cable, for periodic in-situ validation. Integrated sensors enable monitoring of the HVPPS output current and capacitor-bank voltage during each firing event, allowing observation of gradual changes in HVPPS performance. An output filter in the firing path was added to limit the output-voltage transient rise time, protecting the FPV coil from excessive dV/dt slewing and associated insulation damage.

Preliminary results from engineering prototype testing indicate the success of these enhancements towards ensuring a high-reliability and flexible system with integrated sensing and control. Details of the system design, optimization, and fabrication are provided including preliminary laboratory measurements.

Sizing the STEP Prototype Powerplant

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The Spherical Tokamak for Energy Production (STEP) programme aims to deliver a UK prototype fusion energy plant, targeting 2040, and a path to commercial viability of fusion. The STEP Prototype Powerplant (SPP) has the objectives to generate 100 MWe of net electrical power, be tritium self-sufficient, demonstrate high-grade heat, and demonstrate a route to commercial levels of plant availability. STEP has adopted a spherical tokamak geometry, to allow it to operate at high β (ratio of the plasma to magnetic pressure), high elongation and high bootstrap fraction, with the goal of producing a reduced radial size in the design. To minimise the radial build, STEP has been designed with a non-inductive flat-top plasma scenario, to reduce the central solenoid's size by restricting its use to just start-up, and dispensing with inboard tritium breeding.

While it has been shown that the primary size constraint in a conventional DEMO-like tokamak is the divertor, for the STEP device with a double-null advanced divertor configuration the inboard build becomes the primary size driver. Within the inboard build there are two key components that drive its size: the inboard Toroidal Field (TF) coil and the shielding it requires. The TF coil layout is driven by the current required within it to deliver a field on the plasma, the structure required to handle the vertical electromagnetic forces and the manufacturability of the coil. This is not a simple relationship, as increasing the magnet size will increase the major radius of the machine that a minimum TF field is required, thus increasing the current in the inner TF coil, that then increases its electromagnetic load and size . This cycle needs to be balanced to find an overall size that meets the objectives. The STEP device is also being designed with remountable joints, meaning that the centre column is not a lifetime component. Shielding must be added that balances cost between making a larger device and not replacing the centre column too often leading to low availability.

In this presentation we will explore the size of the STEP design. We will discuss the limitations of the existing published size and the workflow developed to explore alternative sizes. We will show the impact changing the TF size and shielding thickness has on the plasma before discussing how we have used uncertainty quantification to drive the size selection. We will conclude by presenting the updated STEP design.

Policy and Strategy of Maintenance during Conceptual Design Phase of DEMO Fusion Power Plant Project

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The European Roadmap identifies the need of the DEMO project aiming to demonstrate that an economically competitive Fusion Power Plant (FPP) is possible before the full commercial deployment of fusion at industrial level. To guarantee the long-term economic viability of nuclear fusion this implies, that the integration of physics and all technological systems must enable a reliable and safe DEMO FPP ensuring a challenging 30 % for the required availability. One of the main contributors to the operational availability is guaranteed by the maintenance function. The function of maintenance engineering is first, to make systems more reliable by reducing FPP failure rates and second, to contribute to the reduction of downtimes required for the three main maintainability indicators involved in the availability equation. For any industrial power plant, the operational availability is impacted by downtimes required for preventive maintenance (scheduled) or corrective maintenance (unscheduled failure), supported by the integrated logistics (spare parts, tooling, facility, engineering documentation…). What is the maintenance policy and strategy needed to yearly guarantee the generation of 2628 hours electricity? This paper sheds light on the basic engineering elements of a robust maintenance policy and strategy, considered in the DEMO Concept Design (CD) phase for a FPP that will enable nuclear safety issues to be managed in a way that is compatible with an optimized life cycle cost. On this basis, the role of maintenance engineering is to define the architecture & data required to draw up a preventive maintenance plan. The paper will also specify the limits of the data that can be collected in the CD Phase, such as a contextual maintenance inventory set for a given work maintenance environment area or the degree of system design development for DEMO remote maintenance tools, especially those designed for corrective maintenance scenarios.

Modal analysis of the wave equation for Gyrotron cavities

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In the field of research related to Gyrotron for nuclear fusion applications, the development of its digital twin is crucial for the optimisation, control and predictive maintenance of the device during its operational life. This requires the implementation of a global model capable of simulating the main physical behaviours that occur in the Gyrotron, in order to predict the output power and efficiency of the device. An important aspect of the digital twin is the ability to perform accurate simulations, drastically reducing the computation time to be comparable to the characteristic times of the Gyrotron.

To overcome the inherent complexity of its multi-physical behaviour, the global model is built starting from the selection and modelling of the relevant components of the device. From this perspective, it is possible to focus on the description of each component and its relevant physical behaviour. Thanks to a state-space formulation of each model of the components, the parameters exchanged by the models have been identified, together with their interconnections.

The aim of the overall model is to simulate the multiphysical behaviour by solving the equations characterising each physical component. In order to achieve a fast simulation, the adoption of a reduced-order model of each component is unavoidable, and its development requires the solution of the so-called high-fidelity simulation in order to collect accurate data of different case scenarios.

The model of the resonator or "cavity" is really important because it is responsible for predicting and calculating the generation of the mm-electromagnetic waves of the Gyrotron. Its development requires the solution of the equations describing the interaction between the electrons and the electromagnetic field.

The cavity model requires the study of the dynamics of electrons affected by a magnetic field, coupled with the Maxwell wave equation of the electric field to simulate this interaction. The use of numerical methods is unavoidable due to the nonlinearity of the electron dynamics for the relativistic correction. On the other hand, the field equation describing the electromagnetic wave generated by an electron density current source is a linear Partial Differential Equation. Proprietary numerical tools such as EURIDICE, TWANG or SELFT have been developed for the study of the cavity and solve the two above-mentioned equations.

The aim of this work is therefore to solve the linear PDE of the amplitude field in a cylindrical waveguide with radiative boundary conditions using modal analysis to obtain an analytical solution. This allows to express the amplitude of the field as a superposition of axial modes describing its evolution in time and space. The perturbation theory was used to study the influence of small deviations of the cavity radius in time and space or the perturbation
of the forcing term. An in-house Python code was developed to compute the solution to the problem and verified against established codes from the literature. The solution of the wave field equation is necessary to generate a data benchmark useful for the development of the reduced order cavity model in the perspective of the Gyrotron digital twin.

STEP Limiter Architecture and PFC Concept Design

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The functions of the Limiters in the STEP Prototype Powerplant (SPP) are to withstand the steady-state plasma heat and particle loads while providing wall protection against offnormal plasma disruptions, such as Vertical Displacement Events and Runaway Electrons. These events will deposit significant amounts of plasma energy onto the first wall in the form of heat flux. The shape and location of the Limiters were set in response to these disruptions and are currently positioned on the outboard regions of the tokamak at the upper, lower and at mid planes. The upper and lower Limiters provide 360-degree coverage through a combination of discrete and panel Plasma Facing Components (PFCs), while the midplane Limiters are at four toroidally-spaced discrete locations.

Previous design iterations of the STEP Limiter PFC included pipe-in-pipe helium jet impingement cooling system with a tungsten heavy alloys (WHA) heat sink. Helium was chosen as the coolant for its good heat transfer performance and recoverability in event of a coolant leak. WHA was selected for its ductility at room temperature and capacity to withstand high steady-state heat loads. However, recent investigations revealed concerns that WHA may lose its ductility under irradiation, which is crucial for its proposed use as the pressure-containing heat sink material. Consequently, the decision was made to replace WHA with CuCrZr, which, following the revision of the heat load requirement and the improvement of jet impingement geometry, now meets the target heat load.

The coolant selection was reassessed in comparison to other coolant options used in the reactor, such as heavy water and CO_2, and was confirmed. The risk of corrosion and oxidation from a Loss of Coolant Accident (LOCA) involving these coolants remains too high to meet the Limiter recoverability requirement.

The revised PFC concept now features tungsten tiles on a CuCrZr heat sink with helium jet impingement cooling. The jet arrays are arranged in parallel channels and integrated into larger rectangular plate-type PFC assemblies, aiming to reduce part count and integrate the manifolding function. Ongoing efforts are focused on refining the PFC design space, enhancing heat transfer, establishing a manufacturing route for plate-type assemblies and integrating them into the larger assembly.

Preliminary Design of the Caesium Oven for the ITER Neutral Beam Injector

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The ITER experiment, currently under construction at Cadarache in southern France, will require a considerable amount of additional power injection from its Heating and Current Drive (H&CD) systems in order to sustain its long plasma discharges. Among all of the various systems, Neutral Beam Injectors (NBIs) will account for 50 MW of power over three devices, mediated by fast neutrals at 1 MeV in Deuterium (or 860 keV in Hydrogen).

The required extracted negative ion current at the Plasma Source can only be reached through efficient negative ion generation and extraction, which in this type of NBIs is achieved by homogeneous surface coating of the extraction apertures with a low work-function metal, such as Caesium.

This paper describes the Caesium Oven for ITER NBI, which is a device for the controlled evaporation of Caesium inside the Plasma Source: its general structure is based on the SPIDER experiment oven design, however with the necessary modifications to make it compatible with the more stringent ITER requirements, such as radiation resistance and Remote Handling compatibility.

This new design has been extensively simulated and a prototype has been tested in the CAesium Test Stand (CATS) experimental setup at Consorzio RFX for the characterization of its parameters.

Thermo-mechanical analysis of a dielectric steering mirror for DTT ECRH launchers

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The launching mirror plays a pivotal role in the Electron Cyclotron Resonance Heating (ECRH) system of the Divertor Tokamak Test (DTT) device, by directing microwaves to required locations into the plasma, thereby fulfilling various tasks, from bulk current drive to instabilities control. Due to the presence of strong and varying magnetic fields at launcher location, it is crucial to explore potential use of materials capable of reducing eddy currents and consequently mitigating magnetic loads acting on the mirror. Silicon Carbide, an advanced ceramic characterized by relatively high thermal conductivity but low electrical conductivity, was chosen as a potential candidate. To ensure the feasibility of the mirror, a customized cooling channel with variable cross-section was designed. After a detailed mesh definition process and a turbulence analysis, thermo-fluid-dynamic simulations (CFD) yielded satisfactory results, with temperature and pressure fields within the expected operational range. Subsequently, a comprehensive investigation of thermal stresses was deemed essential. Utilizing Fluid-Structure Interaction (FSI) simulations, stresses arising from both hydraulic forces in the cooling channels and thermal stresses induced by external loads were examined. The structural analysis, conducted using the failure criteria for brittle materials, showed deformation and mechanical stress values below the material's fracture limit. Since the ceramic bulk is transparent to microwaves, the final step was to choose and evaluate a metallic coating thin enough to sufficiently mitigate eddy currents while ensuring adequate microwave reflection. Finally, a transient structural analysis was conducted, various materials were compared, and an analytical assessment of potential crack propagation was conducted. The presented design activity has laid the foundation for the development of an innovative, actively cooled ceramic mirror for ECRH applications.

Status of high-voltage power supply, integrated control system, and gyrotron of KSTAR EC system for plasma heating

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Wednesday Posters 2, Lobdell (Building W20 Room 208), June 25, 2025, 2:00 PM - 3:30 PM

In Korea Superconducting Tokamak Advanced Research (KSTAR), auxiliary heating systems such as electron cyclotron heating & current drive (ECH&CD), neutral beam injection (NBI), and helicon current drive (HCD) are in operation. EC system is currently one of the most important auxiliary heating systems in KSTAR and flexible used for roles such as magneto-hydrodynamic mode stabilization and plasma heating & current drive, and EC wall cleaning (ECWC). KSTAR is planning to install an electron cyclotron (EC) system for a total of 6MW Radiofrequency (RF) output and is currently in progress. We have been adding EC systems continuously since 2013. In a total of five EC systems installed by 2024, four EC systems (EC2-EC5) use the same 105/140 GHz dual-frequency gyrotron and one EC system (EC6) use the 170GHz single frequency gyrotron. These are all gyrotrons developed by Gycom. Finally, a 170Ghz single frequency gyrotron (EC7) developed in Japan is scheduled to be installed in 2025. The EC2-6 gyrotron is a diode type electron gun, while the EC7 is a triode type electron gun. The EC4-7 system uses the same high voltage power supply and integrated control system. The EC4-7 system uses the same high voltage power supply (HVPS) and integrated control system (ECICS). When operating the EC2-6 system, significant issues with noise and control were discovered. Several ECICS upgrades have been performed over the years to address these issues. To reduce malfunction due to noise, the analog part of the communication section between HVPS and ECICS was minimized by changing it to SPI communication. In KSTAR's EC system, an HVPS based on PSM technology is being commonly used in EC systems. The initially applied PSM control technology was found to have a serious issue causing timing delays. To solve these problems, a new PSM control method of the HVPS controller was developed. In addition, the interlock function of ECICS has been upgraded to protect the gyrotron. The plan to install six EC systems will be completed in 2025. This presentation will explain the problems and solutions discovered over the years of operation. Additionally, the current status of the EC system's gyrotron will be explained, along with a detailed description of the HVPS and ECICS, which were carried out over several phases.

A Modular High Heat Flux Facility to Advance the Science and Engineering of Plasma Facing Components

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While steady-state heat fluxes on Plasma Facing Components (PFCs) can be > 10 MW/m2 depending on the confinement concept, transient heat fluxes from large disruptions can deposit greater than 1 GW/m2 in a few milliseconds onto the PFC. This can damage both the Plasma Facing Material (PFM) and the interfaces to actively-cooled heat sinks leading to leaks and damage to the vacuum vessel. In tokamaks, cyclic transients called Edge Localized Modes (ELMs) are another extreme event that impacts PFCs. Unmitigated ELMs can deposit up to 100 MW/m² onto the PFCs repetitively at 10 Hz or greater, leading to thermal cycling and eventually fatigue and failure of the integrated PFC system. Although PFCs are typically designed by analysis, the Finite Element (FE), Computational Fluid Dynamic (CFD) and Magneto-Hydrodynamic (MHD) codes used must be validated with experimental data, and prototypes must be tested under relevant heat fluxes and duration/cycles to validate expected performance, identify failure mechanisms, and establish fatigue and lifetime limits. Such comprehensive design and analysis cannot be accomplished with modeling alone as most models are not well validated in the operating regimes expected for reactor PFCs. In addition, novel manufacturing methods, expected to be critical for PFC designs in the extreme environments of a fusion reactor, require additional levels of testing and validation beyond the design models. The validation and verification of PFC models is not only necessary to design, manufacturer, and operate PFCs in a reactor environment, it is critical to the overall performance of the reactor.

ORNL is proposing the design and construction of a new facility for collaborative High Heat Flux (HHF) testing of PFCs (divertor, limiter and first wall) at reactor relevant conditions. Ultimately, the facility is intended to be modular and accommodate various heating and cooling methods. Initial high incident heat fluxes will be accomplished initially with a 800 kW electron beam gun that can uniformly heat blanket first wall prototypes (~m²) up to 2 MW/m² steady-state with reactor relevant helium cooling. While for divertor PFCs, the electron beam will be used to test ELM-like and disruption-like transients on smaller test articles (~10cm). Two test articles are being designed and additively manufactured (AM) for future HHF testing: 1) helium impingement-cooled, modular jet (HEMJ) PFC with a simplified tungsten AM tile design and 2) dual-coolant, lead-lithium (DCLL), helium cooled blanket channel test article. Cooling for the facility infrastructure will be provided by a dedicated water loop to remove up to 100 kW/m² of vessel heating due to x-ray production from the e-beam interactions with refractory metal targets and provide cooling to the electron beam. While water cooling will be the initial target coolant available as it's required for the facility, a high pressure, high temperature helium loop for test article cooling is being actively designed to provide continuous high pressure helium (≤ 10 MPa) and reactor relevant flow rates (≤ 1 kg/s), but will not be available when the HHF Facility initially comes online in 2026.

Software Framework Model for Power Supply Control System of Fusion Magnet

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With the development of fusion magnet power supply technology, the transition from traditional phase-controlled power supplies to more advanced fully-controlled PWM (Pulse Width Modulation) rectifier power supplies poses challenges for existing power supply control system software frameworks. These frameworks struggle to fully meet the demands of modern fusion magnet power supplies for highly real-time control performance and rapid acquisition and processing of numerous field signals. To address these challenges, this paper proposes an innovative software architecture—a five-layer framework model based on a data adaptation mechanism and a master-slave architecture design. By utilizing object-oriented programming techniques, we have defined a series of software components tailored for complex converter environments, along with the interface specifications for their interaction. These components support key functions including, but not limited to, real-time regulation of the rectification process, immediate response to fault conditions with protective measures, efficient collection and online filtering of relevant data, ensuring synchronized data transmission, and large-scale monitoring of equipment operating conditions for health status assessment. This solution aims to comprehensively cover the requirements for control, safety assurance, and maintenance management of magnet power supplies in next-generation fusion devices. The proposed software architecture has been validated in practical projects, specifically as a core component of the new fully-controlled magnet power supply control system in a significant upgrade project at the Chinese Academy of Sciences. Since its official trial operation began in early 2022, the system has demonstrated exceptional stability and reliability, proving the effectiveness and advanced nature of the proposed method.

Optimized Divertorlets Designs and FreeMHD Simulations Upgrades for Liquid Metal Components in Fusion Facilities

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The next generation of fusion reactors demands innovative plasma-facing components capable of withstanding intense heat loads under strong magnetic fields. The divertorlets concept addresses these needs by segmenting the divertor into multiple smaller recirculating sections that reduce both the liquid metal exposure to the plasma and the risk of overheating. This approach merges the advantages of fast and slow liquid metal flows, operating at low speeds to minimize magnetohydrodynamic (MHD) drag, while maintaining short exposure times to prevent excessive evaporation. Initial experiments with a divertorlets prototype at the Princeton Plasma Physics Laboratory have aligned with analytical models and MHD simulations, showing promise for further development and eventual implementation in fusion facilities such as NSTX-U. Now, the next steps include refining the divertorlets shape and dimensions to optimize flow stability and integrating a cooling system to manage severe heat fluxes. Additive manufacturing will facilitate practical demonstrations of divertorlets devices, with experimental tests using facilities such as LMX-U informing simulations for iterative design improvements. Concurrently, the advancement of these numerical tools is essential for capturing the full complexity of liquid metal behavior in these extreme environments. FreeMHD, which has previously demonstrated robust performance for MHD flows and free surfaces, is now being upgraded to incorporate heat transfer and thermoelectric effects. These enhancements enable more accurate predictions of liquid metal behavior, accounting for not only fluid interactions with strong magnetic fields but also temperature gradients and associated active cooling. Validation of these solver upgrades will involve benchmarking against experimental data, to ensure reliable prediction of critical phenomena, such as free-surface dynamics and heat transfer under fusion-relevant conditions. The combined efforts of experimental divertorlets optimization and FreeMHD upgrades aim to produce a comprehensive solution for liquid metal divertors.

Advances in Liquid Lithium Technologies at the Center for Plasma-Material Interactions (CPMI)

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Liquid lithium as a plasma-facing component (PFC) presents significant advantages over traditional solid PFCs, including improved plasma performance, high heat removal capabilities and the avoidance of failure modes like transient melting. Lithium PFCs are low-Z, exhibit excellent impurity gettering capabilities, and enable access to the low-recycling regime which may be desired for future fusion reactors. This work presents an overview of several research efforts by the Center for Plasma-Material Interactions (CPMI) at the University of Illinois to understand key performance of lithium and tin-lithium alloys and to demonstrate fusion-relevant liquid metal technologies. The Lithium Alloy Vacuum Appliance (LAVA) device has measured the vapor pressure of liquid lithium alloys up to 1200°C to evaluate their suitability as PFC materials. An investigation into tin and tin-lithium (SnLi) eutectic materials reveals the ejection of macroscopic particles from molten surfaces under hydrogen plasma exposure. Initial findings highlight particle ejection dynamics for pure Sn and demonstrate SnLi particle ejection. Ongoing work focuses on hydrogen accumulation within SnLi, which is critical for understanding tritium retention.
To demonstrate fusion relevant pure liquid lithium technologies, the Actively Pumped Open-Surface Lithium LOop (APOLLO) has been constructed with full funding by Tokamak Energy. APOLLO features a flowing liquid lithium PFC with a free surface, shaped by a 3D ordered in-flow mesh inside a magnetic field. The system has access to either a deuterium plasma exposure produced by a 3 kW Electron Cyclotron Resonance (ECR) source or an electron beam capable of delivering heat fluxes up to 10 MW/m^2. The liquid lithium circulates through a free-surface PFC module and then travels from the PFC to the Hydrogen Distillation Column (HyDE). The latter is is designed for removal of hydrogenic species and impurities by thermal treatment at temperatures exceeding 700°C. Preliminary results demonstrate steady-state hydrogen accounting, TEMHD-driven lithium flow validation, resistance to surface dryout, and effective hydrogen removal. Additionally, ultrasound velocimetry is being explored as a novel diagnostic for characterizing the flow profile in open-surface lithium flow. These combined efforts represent significant progress toward the implementation of liquid lithium PFCs in fusion reactors, addressing key material challenges for lithium and tin-lithium alloys, plasma-material interactions, hydrogenic species accounting, and TEMHD flow across an open surface PFC.

Research on Temperature Compensation Method of Hall High Current Sensor for EAST

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The current level of EAST magnet power supply is high (ten thousand Ampere level), and the currents of power supplies need to be changed in real time to meet the plasma control requirements. It also causes certain temperature changes in the environment where the magnet power supply and current sensor are located. The magnetic field measurement element of the Hall current sensor is a semiconductor device. Its properties such as resistivity, mobility, and carrier concentration are temperature dependent. These temperature dependencies cause variations in Hall voltage, sensitivity coefficient, internal resistance, and other parameters, which in turn affects the measurement accuracy of the sensor. To address this issue, effective temperature compensation need to be implemented. Traditional methods such as parallel temperature compensation and constant current source input compensation can compensate for the lower-order errors, but fail to completely address the higher-order errors introduced by temperature variations. Therefore, these methods are not suitable for environments with significant temperature fluctuations. In this article, a new feedback-based temperature compensation method is proposed. This method improves the temperature characteristics of the Hall element by multiplying the compensated output signal with the temperature compensation signal and then adding the resulting product to the Hall element's output signal. This approach further enhances the temperature performance of the Hall element and improves the environmental adaptability of the Hall current sensor. Simulation and experimental results demonstrate that the feedback-based temperature compensation method effectively eliminates the impact of temperature changes on the Hall element parameters, and significantly improve the measurement accuracy of the sensor, with the measurement error controlled within 0.4%.

Design and Commissioning of ELIOS Facility for Materials Compatibility in Lithium Loops

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The Experimental facility on LIthium cOmpatibility with materialS (ELIOS), which is located in the Liquid Metal Laboratory at LNF-CIEMAT, has been designed to test the compatibility of different materials, weldings and gaskets working in a liquid lithium environment, particularly in lithium circuits related to the fusion field. Key examples of these circuits include the Lithium Loop and Purification System of DONES and the LITEC (Lithium Technologies CIEMAT) experiment.
ELIOS features a modular design with various inert-gas-filled hot boxes for added safety, maintaining a non-reactive atmosphere for potential lithium leaks. Each of these hot boxes contains crucibles holding liquid lithium. Certain crucibles are designed to enclose sample test materials, whereas others are sealed with gaskets to be tested. These crucibles are exposed to controlled pressure and temperature for extended periods managed by SCADA (Supervisory, Control and Data Acquisition) system. Besides, Online monitoring detects seal failures in sealed crucibles by analyzing subtle pressure fluctuations.
ELIOS provides online monitoring and simulates realistic operational conditions where molten lithium interacts with candidate samples and gaskets. After exposure, the samples are extracted for detailed analysis to evaluate chemical degradation or structural damage. Also, the assessment of gasket performance is conducted through analysis of their operational lifetime. This work represents a step forward in improving the reliability and safety of advanced lithium systems for fusion technology.
Keywords: Liquid metal, Compatibility, Fusion materials, IFMIF-DONES, LITEC

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A perspective on how MPEX will serve the US materials roadmap

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The US program is in the midst of a strategic long-range planning exercise, including making roadmaps for critical fusion technologies. Foremost is the development of fusion materials able to withstand the harsh conditions of a fusion pilot plant. Since the strong emerging fusion industry time scales demand a faster development track for fusion materials. In this contribution the development of plasma facing material and components is in focus. As part of the screening process new plasma facing materials need to be exposed to fusion grade plasma conditions. The Material Plasma Exposure eXperiment (MPEX) is a new linear plasma device in construction at ORNL, which will fulfill this role. MPEX will be able to re-create the plasma conditions (electron temperature, electron density, ion flux) expected in a divertor of a toroidal magnetic fusion reactor. In addition, it will be able to handle hazardous materials, such as alkaline metal targets and neutron-irradiated materials. MPEX will be a superconducting device with a total of 1 MW of heating power. It will have the capability to transfer targets from the plasma generator to a surface analysis station with state-of-the-art surface diagnostics intermittently such that the evolution of surfaces in very long pulses can be documented without extraction of the sample.
As the first MPEX research program is being developed, the need of the US and international community is taken into account. First experiments will be informed by the materials roadmap and will drive the emphasize in the high-level-commissioning phase.

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Design of a PbLi Corrosion Testing Loop and Experimental Planning

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Lead-lithium (PbLi) is the worldwide front-runner for liquid tritium breeder material of fusion reactors. Corrosion of structural blanket materials such as Reduced Activation Ferritic Martensitic (RAFM) steel in the flowing hot PbLi in the presence of a strong plasma-confining magnetic field is among the most serious limitations in the design of a robust blanket system. A new flowing PbLi corrosion loop with a magnetic field and surface heating in the test section, reaching prototypical blanket conditions is under development at Oak Ridge National Laboratory. The proposed PbLi loop will be able to show the corrosion performance of candidate fusion materials in an experimental fusion prototypical environment and establish their corrosion mitigation. The developed computational tools are used in pre-experimental simulations to carefully plan future experiments on corrosion/precipitation to deliver data for validation of phenomenological models and computer codes and help in the design and analysis of real blanket systems.

Deployment and validation of the SOPHIA tokamak simulator on multiple platforms: ST40, SMART, TRUST and FPP

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SOPHIA is a versatile tokamak plasma simulator, developed by the UK-based private fusion energy company Tokamak Energy Ltd. It integrates models for the plasma, diagnostics, actuators and the plasma control system, providing a ‘control room’-grade experience to the operator.

SOPHIA has been successfully deployed on the high-field spherical tokamak ST40 in the preparation of new scenarios (for both single and double-null diverted configurations), in the validation of experiments, the testing of new controllers and in the training of session leaders. The validation of SOPHIA and its physics models on ST40 enables the development of plasma scenarios in the next generation of devices, such as a spherical tokamak Fusion Pilot Plant (FPP: Tokamak Energy, supported by the US DOE Milestone-Based Fusion Development Program), to be carried out with a higher level of confidence. SOPHIA can be used to test whether a proposed design point is feasible by assessing the controllability of the design under a set of assumed realistic models for the control system, device actuators, diagnostics and plasma dynamics.

The integration of SOPHIA with existing pulse preparation and analysis tools makes it ideal for the training of new ST40 pilots and session leaders. Old pulses which had been executed with pulse preparation errors, as well as newly generated errors, can both be added to SOPHIA runs, enabling a pulse database to be constructed, from which AI based virtual pilots for future fusion plants could be trained.

By design, SOPHIA is a machine-independent tool that can be deployed on other tokamaks. In this work, we will describe the implementation of SOPHIA on the SMART (University of Seville) and TRUST (University of Tuscia) tokamaks, as well as on Tokamak Energy’s FPP, and report on the results of predictive simulations. Using the MATLAB/Simulink environment, SOPHIA integrates the transport code ASTRA, which is tightly coupled to the equilibrium code SPIDER, as well as models for diagnostics, actuators and the control system. In the case of modelling for the TRUST tokamak, SPIDER is replaced by an evolutionary equilibrium code that was developed in-house at the University of Tuscia, supported by external MATLAB/Python external routines.

ITER Electron Cyclotron Heating System: Confinement Strategy and Qualification Program for Diamond Window Unit and RF UHV Isolation Valve

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The Electron Cyclotron Heating (ECH) system of ITER shall provide microwave heating and current drive to the ITER Plasma. In this sense, it is aimed at providing local current drive to stabilize Neoclassical Tearing Modes (NTMs), assist in the plasma breakdown and burn-through and contribute to the plasma heating.

From a nuclear safety perspective, its function is to contribute to the confinement of radioactive materials. Two components are critical in this aspect: a Chemical Vapor Deposition (CVD) Diamond Window Unit (DWU) and an All-metal RF UHV Isolation Valve (IV). Together they form the last barrier of the First Confinement System (FCS) of the ECH system. These components shall be designed and manufactured to ensure confinement of tritium and dust while enabling the passage of millimeter radiofrequency waves for plasma heating.

The confinement strategy applied relies on the performance and complementary roles of both components. They can provide physical separation between the tokamak vacuum and the Transmission Line (TL) vacuum. The DWU is a passive barrier while the IV is an active component, seated between the Vacuum Vessel (VV) and the DWU. When open, it allows the passage of microwaves, when closed, it becomes the separation point of the FCS.

Fusion for Energy (F4E), the European Domestic Agency (DA) for the ITER project, is responsible for delivering an Electron Cyclotron Heating (ECH) system that can both perform functionally and meet the safety requirements.

A Qualification Program is being established to demonstrate the safety performance of the two critical components. Their qualification will be achieved through a combination of analysis and testing:

- Qualification by Analysis. Finite Element (FE) models of the components will be developed. Their structural integrity will be assessed against the rules of the RCC-MRx Ed. 2022. Typically, pressure, thermal, mechanical and inertial loads will be evaluated. A Qualification by Similarity approach will be used to assess the effect of radiation and magnetic loads.
- Qualification by Testing. The RF valve closing mechanism, along with certain design aspects of the DWU, are too complex to be easily evaluated through FE analysis, which would lead to significant uncertainties in the results. For this reason, both components will undergo a testing campaign where their safety performance (e.g. safety leak rate and operability) will be assessed.

This paper outlines the chosen confinement strategy, including redundancy and diversity, a description of the two physical barriers and details the specifics of the Nuclear Qualification Program, including the criteria, qualification requirements, methods, results achieved and upcoming stages.

Speed is even more important than we thought: optimizing fusion plant maintenance schedules to maximize their value in a decarbonized electricity system

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With optimized scheduling, performing maintenance in multiple short blocks rather than one long block can increase the value of a power plant even as the availability decreases. In our model of a future decarbonized US Eastern Interconnection, given a plant with a five-year environmental cycle, if one could replace a component such as a blanket over four 16-week outages scheduled in springtime rather than one year-long outage, the break-even cost of the power plant could increase by 8% even as the availability drops from 0.80 to 0.76. As a further example, by performing maintenance in spring, when the large capacity of variable renewables drives down the price of electricity, a plant with a 16-week maintenance outage every year retains about 84% of the value of a maintenance-free plant, not 70% as would be expected from its availability. Fusion plants can be designed to provide electricity when it is most needed, not necessarily year-round.

If faster maintenance could be accomplished, at the right time of year, components could be designed with lower durability requirements, easing engineering demands on the structures and the required timelines for materials qualification. A plant with a \$100M, 1.5 full-power-year (FPY) blanket which is replaced every other year in an 18-week block would be similarly valuable to plant with a \$200M, 4 FPY blanket replaced every 5 years during a 38-week block, even though its overall availability is lower and the annual blanket replacement cost is higher.

The optimal design of a plant and its maintenance cycle depends on the electricity system it will be situated in. We propose a simple method to choose an optimal fusion plant maintenance schedule, given an electricity price series and a few generic maintenance-related parameters; this is valid in a perturbative regime with few fusion plants or infrequent maintenance. To test this model, we perform self-consistent simulations of a future decarbonized United States Eastern Interconnection power grid with a large fleet of fusion plants, optimizing their maintenance scheduling. The results agree except for cycles with maintenance every year, where the perturbative model overestimates the plant's value.

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Real-time transmission and analysis of Hypersim fusion power simulation data based on the UDP protocol

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As the complexity of modern power systems increases, the use of real-time simulation tools such as Hypersim in power system research and analysis becomes increasingly important. However, how to efficiently and accurately transfer Hypersim simulation data to external analysis tools (e.g., MATLAB) is a key issue. In this study, we design and implement a data transfer architecture based on the UDP protocol for real-time transfer of electrical signal data from Hypersim to external analysis platforms.The UDP protocol has been selected as the core communication protocol for data transfer due to its low-latency characteristics.

In this paper, firstly, the generation mechanism of Hypersim simulation data and its importance in real-time power system simulation are described. Then, the implementation process of the UDP communication protocol is described in detail, including the design of the data packing format, the key algorithm of data parsing, and how to efficiently collect and process the received data in the MATLAB environment. The experimental part verifies the performance of the proposed method: even under high-frequency data transmission scenarios (e.g., 20 kHz sampling rate), the system is still able to achieve stable real-time data stream reception and electrical signal reconstruction.

The experimental results show that this method can complete the real-time transmission and accurate restoration of Hypersim simulation signals in an efficient and low-latency manner, which provides reliable technical support for future applications such as optimal control and fault diagnosis based on real-time simulation signals in power systems. The results of this study will further promote the practical application of real-time simulation technology in complex fusion power sources.

Development of ceramic coatings using PVD and PA-CVD techniques for tritium permeation barrier applications

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The developing fusion energy sector necessitates the utilization of materials able to sustain extreme conditions and harsh environments. Materials that are planned to be used in various areas of fusion reactors are presented with manifold challenges. Required properties include, amongst other, heat resistance, high-temperature corrosion resistance towards liquid metal coolants, dielectric properties to minimize magneto-hydrodynamic effects, stability under radiation and high-fluence ion/neutron bombardment, as well as barrier properties towards hydrogen isotopes. Structural materials do not simultaneously fulfil all the stringent requirements above, hence the ambition of providing them with additional properties that could enhance their performances in such an extreme environment. A popular strategy consists in applying multi-functional coatings, given the flexibility in tailoring the surface/material properties that this method allows. In this context, different property requirements could be achieved using a multi-layer approach, where each layer tackles different challenges. Well established techniques for applying coatings include Physical Vapour Deposition (PVD) and Chemical Vapour Deposition (CVD) methods, potentially with plasma assistance (PA-CVD).
Ceramic coatings, in particular, have attracted interest due to their potential as tritium permeation barriers, feature that is crucial for the management of the tritium fuel cycle in fusion reactors.

In this work, the deposition of ceramic coatings via PVD and PA-CVD, carrying additional functionalities, is studied. The flexibility given by the magnetron sputtering deposition technique allows for a fine tuning of the stoichiometry of the coatings. This includes the possibility of adding dopants, to favour the formation of a certain crystalline structure or an amorphous compound. In this context, metal oxides, including binary and ternary rare-earth metal oxide systems, have been deposited on coupons of structural grade materials, using varying levels and types of dopants. Moreover, in order to promote the adhesion of the deposited layers, a study on possible interlaying materials has been carried out. The produced coatings have undergone testing aiming at characterizing their adhesion strength and their crystalline/amorphous nature, both in their as-deposited state and after an in-vacuum heat treatment. Best performing specimens have undergone ion irradiation and exposure to tritium, and have been subsequently characterized further to evaluate their stability and their barrier properties. Main results of this materials study will be presented.

Any scale-up endeavour aiming at industrializing the application of such ceramic materials on the structure of fusion reactors would require the coatings to be deposited on irregularly shaped parts. The pipes carrying the liquid lithium, for example, would need to be coated internally, for a tritium fuel cycle to take place efficiently. To accommodate the internal coating of tubular parts, we have developed two different coaters, based on PVD and PA-CVD techniques, designed for tubing and piping with different internal diameters. In each case, their inherent challenges will be presented, together with characterization results acquired on selected oxide layers deposited in both manners.

Thermal Analysis and Design Optimization of DC Link for Long Pulse Operation in the BEST Tokamak

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The BEST is a Tokamak fusion device for achieving D-T (deuterium-tritium) burning, where the DC Link serves as the critical electrical circuit for delivering current and voltage from the magnet power supply to the superconducting coil. This function is essential for controlling plasma shape, position, and providing ohmic heating. Long-pulse operation of Tokamak devices demands precise thermal management to ensure stable performance, making it vital to regulate the operating temperature of DC Link within a defined range. In this paper, the thermal equilibrium analysis of DC Link designed for a rated current of 55 kA is presented. Utilizing DC Link's low inductance and high insulation properties, the positive and negative conductors are assembled in insulating layers and metallic casing. The temperature distribution within the conductors and insulating materials under two cooling scenarios is analysed: without cooling water and with water cooling for long-pulse operation. Additionally, the impact of contact resistance on the temperature rise at the junctions of the multi-terminal busbar is calculated. The relationship between busbar length, conductor, metallic casing and insulating material temperatures, and flow velocity is analysed by the finite-element method. The maximum operating time of the DC busbars without cooling water is analysed to ensure the reliability of the DC Link system. This work provides critical insights into the thermal management of high-current DC Link systems for superconducting magnets in long-pulse Tokamak operations.

Status of the Progress of Fabrication of High Power Ion Extractor Grid for NBI System

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Neutral Beam Injection (NBI) is the workhorse for heating and current drive in Tokamak fusion plasma worldwide. The heart of the NBI system is ion extractor grids which play a vital role in extracting ions from the plasma source and accelerating to the desired energy. Steady State Superconducting Tokamak (SST) has a provision of a positive ion based NBI system for delivering 1.7 MW neutral hydrogen beam power at 55 kV. To fulfil this requirement, the extracted ion beam power is 5 MW and the extracted hydrogen ion beam current is 90A. To obtain such a high ion current from the source, the ion extraction area must be large enough to accommodate a few hundred shaped apertures. SST NBI system has 3 grid accel-decal system e.g. Acceleration Grid (at 55 kV), Deceleration Grid (-2 kV) and Earth Grid (at zero voltage), each grid has 774 shaped apertures drilled on Oxygen Free Electronic (OFE) copper plate. The shaping of the aperture is required for the generation of a low divergence beam and designed by ion beam optics simulation using the AXCEL-INP computer program. Focusing on such a large number of ion beams is done by mechanical offset (400 µm max.) between the grid apertures. Horizontal and vertical focal lengths of the SST NBI beam line are 5.4 m and 7 m respectively. During beam operation, the grid received a heat load of 1.75 MW/m2 which is removed by a dense network of water cooling channels embedded between the rows of apertures. The ion optical design demands stringent tolerances in various dimensions e.g., the position of the aperture is 40 µm, the radius of the wavy semi-circular cooling channel is 50 µm, the surface flatness is 100 µm, the diameter of the reference hole is 10 µm and thickness is 50 µm etc. Complex technology and techniques are involved to achieve such critical tolerances during fabrication. As an in-house technology development program, ion extractor grid fabrication routes and technology are identified and understood based on the results of prototype test samples. There are 3 technologies involved e.g. (i) joining of SS304L rod to OFE copper plate by Friction Welding (FW) for fabrication of header pipe to supply the water to the cooling channels (ii) copper electro-deposition for making embedded cooling channels inside OFE copper base plate (iii) special fixtures for CNC machining of OFE copper base plate to obtain the above mentioned specified dimension tolerances. This paper shall describe the status of the progress of the ion extractor grid fabrication work. The entire work is divided into two phases, the deliverable items in Phase-1 are one half-size Prototype Acceleration Grid (PAG) with 7 Nos. of fixtures. In Phase-2, the deliverable items are the actual size of (i) Acceleration Grid (2 halves) and 7 Nos. of fixtures (ii) Deceleration Grid (2 halves) and 4 Nos. of fixtures (iii) the Earth Grid (2 halves) and 3 fixtures. This technology development work will help the Indian Tokamak fusion program.

Fusion Power Plant Design and Optimization with the FUsion Synthesis Engine (FUSE)

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The FUsion Synthesis Engine (FUSE) [1] is an open-source software suite developed by General Atomics to guide the design and optimization of fusion power plants. FUSE's modular framework supports a hierarchy of fidelities, from high-level system codes to high-fidelity physics and engineering models, enabling a comprehensive design workflow from pre-conceptualization through to detailed verification and validation. By integrating model outputs in a global IMAS data dictionary, FUSE is used to generate self-consistent solutions for all major power plant subsystems.

We present results for the FUSE optimization of a net-electric, high-field tokamak fusion power plant, wherein a pareto-optimal population of solutions is found that demonstrates the trade space between design risk (i.e. proximity to operational limits) and project cost (i.e. capital and O&M expenses). This particular strength of the FUSE code uses rapid design space exploration in which tens of thousands of device solutions are generated and compared through the use of parallel computing and multi-objective optimization. Physics and engineering constraints are applied with varying safety margins to determine design risk, and the sensitivity of FUSE results to these constraints is explored and analyzed.

This approach to device design is shown to yield two important types of result. First, the pareto-optimal technique allows the user to conduct risk-cost-benefit analysis on a range of solutions, and to select a design path tailored to stakeholder needs. Second, the method of constraint analysis identifies which technological capabilities are most leveraging in power plant design, and informs the critical steps on the technology development roadmap toward fusion energy.

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[1] Meneghini, O., et al. FUSE (Fusion Synthesis Engine): A Next Generation Framework for Integrated Design of Fusion Pilot Plants. https://doi.org/10.48550/arXiv.2409.05894.

The SPIDER Power Supplies towards full beam source performance

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The SPIDER experiment is in operation at the Neutral Beam Test Facility in Padova, Italy, to demonstrate the performance requested for ITER Neutral Beam Injectors in terms of extracted ion current density and beam divergence in a Radio Frequency (RF) ion source in vacuum. The beam is produced from a hydrogen plasma and accelerated up to 100keV. The SPIDER Power Supply system is composed of two main subsystems: the Ion Source and Extraction Power Supplies (ISEPS), devoted to produce and extract the RF plasma, and the Acceleration Grid Power Supply (AGPS), which is aimed at accelerating the beam up to -100kV. The ISEPS is hosted in a Faraday cage insulated with respect to ground, called High Voltage Deck (HVD) and negatively polarized by the AGPS. In 2022SPIDER went into a long shutdown, following the first extensive campaign in Caesium, to improve the beam source as a result of the troubleshooting and the experience gained in the first phase of the operational life. In particular, modifications have been implemented in the RF drivers to improve voltage withstanding capability and matching, and in the RF connection path to reduce the mutual coupling among different circuits. SPIDER went back into operation in 2024 for a long Caesium campaign until early 2025.
From the point of view of the power supply system, two main issues emerged from the first operational phase of SPIDER. The first one was related to the resilience of the power supplies to repetitive breakdowns; such events, i.e. the sudden short circuit to ground of the extraction grid or between the extraction grid and the plasma grid, even though they have to be considered as a normal operating scenario for the device and must be tolerated without stopping the system, gave rise to fast transients resulting in overvoltage, overcurrent and EMI impacting the power supplies, which ultimately led to a limitation of the operational parameters of SPIDER. Some countermeasures were adopted in the past already, but were not enough to enable operation at full acceleration voltage. The second issue was related to the use of RF oscillators, which intrinsically prevented the achievement of a perfect matching of the driver loads due to the flip phenomena, therefore limiting the maximum power that could be transferred to the beam source plasma. Again, although an optimization and fine tuning of the system were performed and resulted in a strong increase of the performance, the achievement of the nominal 200kW per generator power was still not possible.
This paper discusses from an historical perspective the modifications, developments and upgrades to the power supply system and RF circuits which were necessary to support the operation the SPIDER beam source in the last CAasium campaign and beyond; the lessons learned from years of operation and experience will be summarized describing the steps that made possible for the system to finally achieve high voltage operation in presence of repetitive breakdowns and high RF power delivered to the plasma. Pending limitations to the power supply system and to the beam source drivers and circuits will also be highlighted in view of the operation with four beam segments, with a discussion on possible proposed solutions.

Controllability Analysis of Electromagnetic Field Profile in Tokamak Plasmas Using Port-Hamiltonian System

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The realization of sustainable nuclear fusion power requires advanced methods to control the dynamic behavior of tokamak plasmas. These plasmas feature is highly complex characteristics due to nonlinearities, distributed parameters, and interactions across multiple physical domains, such as thermodynamics, electromagnetism, and fluid dynamics. To address these challenges, robust modeling frameworks and innovative control strategies are essential.

This research adopts the port-Hamiltonian (PH) system framework, a powerful modeling approach that can naturally incorporate the nonlinear and distributed characteristics of physical systems. The PH framework is particularly suited for tokamak plasma dynamics because it describes systems through energy-based interactions, enabling a unified representation of multiple physical domains. By defining systems with structure-preserving equations, it can simultaneously account for energy storage, dissipation, and exchange between domains like heat transfer, electromagnetic fields, and fluid flow. This makes the PH framework an ideal choice for analyzing and controlling complex, multi-physics systems such as tokamak plasmas.

Using this framework, we analyze the controllability of the electromagnetic field in a 0D thermo-magneto-hydrodynamic (TMHD) model. This model captures the essential dynamics of electromagnetic interactions within tokamak plasmas, allowing us to investigate whether the spatial distribution of the electromagnetic field can be manipulated as desired using available control inputs. In this context, controllability refers to the ability to adjust the spatial distribution of the electromagnetic field through control inputs. By quantitatively evaluating the controllability of this model, we aim to identify potential limitations and opportunities for improving actuator design and control strategies in fusion reactors.

Our first trial reveals that the spatial distribution of the electromagnetic field is uncontrollable under certain limited conditions. Specifically, it was quantitatively demonstrated that the current actuator configurations are insufficient for intentionally controlling the electromagnetic field distribution. This highlights not only the limitations of existing actuator designs and placements but also the necessity for selecting or designing new actuators to address these shortcomings. Moreover, this study serves as quantitative and fundamental basis for the development of advanced fusion control systems, providing critical insights and benchmarks that can guide future innovations in plasma control strategies and actuator technologies.

Recent progress in PPPL experimental liquid metal program: design of LEAP and insights from LMX-U

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Flowing liquid lithium plasma-facing components (PFCs) offer transformative benefits for fusion devices, including enhanced plasma stability, improved heat and impurity management, and reduced reactor size through low-recycling regimes. However, challenges such as lithium safety, flow stability, and practical implementation remain. To address these, the Lithium Experimental Application Platform (LEAP) was developed as a 3m x 3m x 2m modular gloveroom system operated in an argon environment, supporting up to a 50 lb (22.68 kg) liquid lithium inventory with heating and strong magnetic field capabilities. LEAP enables the testing and prototyping of liquid lithium PFCs. Initial validation experiments on the LMX-U device using GaInSn have provided insights into the stability of free-surface liquid metal flows, informing the integration of a free-surface liquid lithium module into LEAP and advancing the adoption of liquid lithium technologies in fusion reactors.

Engineering and physics challenges in high-power operating scenarios in the Spherical Tokamak Advanced Reactor (STAR)

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The Spherical Tokamak Advanced Reactor (STAR) fusion power plant design study is exploring physics and engineering integration issues for a low-aspect-ratio steady-state tokamak-based fusion power plant with net-electric power output in the range of 100-1000MWe. STAR design study parameter ranges are aspect ratio A = 2, R0 = 4-4.5m, BT = 4.5-5.5T, elongation κ = 2.25-2.5, and triangularity δ =0.6-0.65. Present studies are exploring the implications of pushing the overall core plant size and power generation capabilities toward Gen III AP1000 fission-based power plants. Net-electric power generation at 1GWe (Pfusion = 2-2.5GW) in a R0=4.5m device with 30 year central toroidal field (TF) magnet lifetime (determined by neutron fluence limits) appears feasible, but would require advanced physics and engineering on several fronts. Operation above the no-wall stability limit at normalized beta βN = 4.5-5 and with 80-90% bootstrap fraction would be required while retaining the favorable confinement scaling observed in present ST devices NSTX/NSTX-U and MAST/MAST-U. Power exhaust at these power levels is very challenging and is motivating extensive studies of liquid metals (LM) – in particular a lithium vapor box divertor combined with noble gas radiative power dissipation simulated using SOLPS-ITER. The compact divertor of STAR requires integration with the shielded poloidal field (PF) magnet system, and pushes PF coil current densities to high values approaching the values needed for the TF coils and central solenoid. The average neutron wall loading in this scenario is 2-3MW/m² with high peak outboard values approaching 5MW/m² which will challenge the material limits and power removal capabilities of the dual-coolant lead-lithium blanket under consideration. These and other engineering challenges will be described. Comparisons to higher aspect ratio steady-state tokamak projections will also be discussed.

The Preliminary Design of an LD-FIRST (Laser Driven Fusion Integration Research and Science Test Facility) Informed by Integrated Systems Modeling

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A comprehensive integration and test facility, the Laser Driven Fusion Integration Research and Science Test Facility (LD-FIRST), is proposed to serve as a full-scale community testbed for essential technologies across Inertial Fusion Energy (IFE) and Magnetic Fusion Energy (MFE) concepts. Building upon the singular achievement of fusion ignition and scientific breakeven at the National Ignition Facility (NIF), LD-FIRST aims to leverage decades of research in IFE, including the LLNL Laser Inertial Fusion Energy (LIFE) study.

Designed to operate with deuterium-tritium fuel, LD-FIRST is expected to generate plant-level 14.1 MeV neutron output, facilitating comprehensive materials and performance testing of fusion blankets and fuel cycle systems, including tritium breeding and recovery. The facility will also accelerate the development of foundational component and system technologies for the fusion community. By utilizing the point source nature of Inertial Confinement Fusion (ICF) and the modular capabilities of IFE, LD-FIRST can function as an accelerated testbed for various blanket module concepts pertinent to both IFE and MFE, with adjustable irradiation fluxes based on proximity to the target chamber's center.

Achieving yearly displacements-per-atom (DPA) of 10 or higher for meter-scale objects is feasible even at a yield of just 10 MJ per shot (a near-term NIF goal, with a 5.2 MJ record yield achieved in 2024), contingent upon achieving IFE-scale repetition rates of around 10 Hz. LD-FIRST represents a critical risk-mitigation strategy for fusion power development, fostering diverse research pathways in synergy with other integrated demonstration facilities, such as ITER.

Currently in the pre-conceptual design phase, LD-FIRST will draw extensively from the substantial findings of the LIFE study, which itself is built upon a rich history of prior IFE system design efforts. The facility will employ an indirect-drive scheme, with a driver/target physics foundation validated by NIF experiments. An internally developed system model, the Integrated Process Model (IPM), will be leveraged to optimize LD-FIRST's design, considering over 100 user input parameters and performing a self-consistent techno-economic assessment of facility architecture.

Successful realization of LD-FIRST will necessitate robust partnerships across the IFE, MFE, and ICF communities, as well as collaboration with public and private sectors and their supply chain vendors. These partnerships will be vital in defining the detailed design of and pathways for LD-FIRST development. The timely advancement and deployment of LD-FIRST will leverage the United States' historical leadership in fusion science, propelling the nation towards achieving fusion commercialization and ensuring energy security.

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Design and Research of Magnetic Self Moving Bolt Disassembly and Assembly Robot for DEMO Remote Maintenance Strategy

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The DEMO remote maintenance strategy includes a necessary task to efficiently and accurately complete the disassembly and installation of bolts. If a traditional robotic arm is used to disassemble and assemble one by one, the efficiency is very low, and it is difficult to achieve accurate bolt pretensioning.
Therefore, this study intends to design a remote automatic magnetic adsorption wall-climbing robot, which can carry out construction operations on vertical door panels. It is equipped with a vision sensor to achieve accurate bolt positioning and bolt model confirmation. The wall-climbing robot is also equipped with two sets of hydraulic bolts pulling systems and torque detection sensors, which can work together to achieve precise preload application of bolts, thus completing the automatic tightening and disassembly of bolts.
The core technical solutions of this study include: the layout design and optimization of the whole machine scheme to ensure the feasibility of the narrow working space; calculation, analysis and optimization of magnetic attraction, friction and walking driving force of wall-climbing robots; structural design and optimization of hydraulic bolt tensioner; innovative design and analysis and verification of variable pitch mechanism.
The design and development of the equipment can efficiently and accurately realize the disassembly and assembly of large door panel threaded fasteners to ensure the smooth implementation of remote operation and maintenance operations.

Identifying reduced-order models from high-fidelity simulations for density control with pellet injection

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Predictive high-fidelity simulations serve as testbeds to validate and develop control strategies for actual reactors, but their complexity limits real-time use for control. This work explores methods to identify reduced-order models from high-fidelity JINTRAC simulations for ITER. It focuses on density control with pellets to enable real-time predictive control and assess their applicability to a reactor.

Pellet injectors are the fueling actuators that launch high-velocity fuel pellets into the edge of the core plasma. The injection poses the risk of exceeding the edge stability limit, which the density controller must address. Future reactors are expected to use predictive control methods to manage plasma density while adhering to stability constraints during pellet injection. These controllers will rely on observers reconstructing the density profile from diagnostics in real-time.

The effectiveness of predictive control and observer solutions relies on the quality of the predictive plasma density models used. Therefore, developing control-oriented density models with pellet injection is crucial for the safe operation of a fusion power plant.

This work includes identification methods that use a black-box method, Dynamic Mode Decomposition for Control (DMDc), and two grey-box methods: physics-informed Dynamic Mode Decomposition for Control (piDMDc) and estimation of a partial differential equation with spatially varying transport parameters. Dynamic data for identification is obtained through periodic pellet frequency perturbations that induce a global response in the plasma density. These perturbations are initially tested on a drift-diffusion particle transport model to validate the approach and methods.

As a next step, pellet perturbations are simulated in a high-fidelity JINTRAC simulation for ITER. From this simulation, both DMDc and piDMDc methods identify a reduced model comprising only a few states while accurately predicting the evolution of plasma density with pellet injection. However, identifying spatially varying transport parameters has not yielded reliable results and requires further investigation.

The piDMDc method is preferred over DMDc due to its higher accuracy, lower model order, and flexibility in imposing stability and other constraints. Future work will focus on improving the identification of transport parameters for a more interpretable and adaptable model.

In conclusion, this work identifies reduced-order models for density control using pellet injection. We validated the identification methods, showing their accuracy in predicting density evolution in JINTRAC simulations for ITER. These models can be used in predictive density controllers and observer design. Furthermore, these methods can be applied to reactors to generate data for model identification and validation.

HTS Power Supplies for DC Fusion Magnets

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HTS Power Supplies for DC Fusion Magnets

OpenStar is building a levitating dipole reactor (LDR), following in the footsteps of LDX and RT-1. In our quest to achieve fusion we are building Junior; an assembly of non-insulated (NI) high temperature superconducting (HTS) pancake coils weighing 500 kg operating at 1.44 kA and a peak field of 5.6 T.

An enabling step for the LDR is developing an integrated power supply system which has the ability to maintain the current in the magnet while it is levitating and confining plasma. Recent developments in HTS superconducting power supplies have shown that they are a viable solution to keeping an HTS magnet charged with minimal heat leak to the magnet (operating at 30-50 K). We will detail the challenges of building and operating a power supply which needs to interface with two drastically different temperature regimes (50 K and 300 K) in addition to another intermediate regime (65 K).

While levitated, we need the power supply system to operate without supervision, maintaining the current and watching for crucial events; like over temperature, which can be communicated via simple IR communications. A traditional suite of semi-conducting power supplies and control electronics were developed and integrated which is housed in an internal cavity on-board Junior.

In this presentation we will present the performance specifications of the developed HTS power supply system, such as charge time, energy efficiency, and thermal performance of whole dipole system during our first plasma experiment.

Improved Technology Readiness Assessment Method for Fusion Remote Maintenance Systems from a Product-Driven Design Approach Based on Design by Cataloguing

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The design approach for remote maintenance (RM) systems for EU-DEMO is now focused on defining equipment with appropriate maturity and technology readiness level (TRL) to shorten the lifecycle and engineering design process while implicating existing and matured technologies in the fusion engineering design. A novel product-driven (PD) approach, based on the design by cataloguing, has been introduced to define mature equipment with higher technology levels. Therefore, the PD is a COTS-driven design approach characterized as the further reduction of technology risk, primarily related to integrating COTS technologies into the fusion tokamak system design. The PD activities include the continued maturation of COTS systems (concept maturation (CM)), integration activities to ensure compatibility in the DEMO tokamak environment and testing to ensure performance and reliability. Technology readiness assessment (TRA) is a crucial tool in CM for quantitatively evaluating COTS system technological levels (STL), judging the feasibility of application in DEMO RM, and managing the associated risks. However, since PD is contextualized as an integrated complex system, the current TRA, which relies on the TRL method, which is suitable for assessing individual technologies and has limitations in evaluating SRL, is inadequate. An Improved Technology Readiness Assessment (ITRA) is proposed as a method for the PD design. The ITRA improves the TRA framework by refining TRA procedures and criteria from a systems-integration perspective. The proposed ITRA framework incorporates the EU-DEMO Tokamak's operational capabilities, technical difficulties, and critical RM technologies in the assessment. The proposed method is demonstrated through the ongoing development activities for the Tokamak Ports and In-BioShield RM equipment. The preliminary results showed improvements in the assessment and synthesis of complex systems' design by systematically reviewing the TRL of available commercial off-the-shelf (COTS) solutions while performing functional analysis and evaluating each solution's technology maturity.

The Spherical Tokamak Advanced Reactor (STAR) fusion power plant design

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Many proposed magnetic fusion DEMO and compact pilot plant designs are extensions of existing physics focused experimental devices defined to understand and control plasma operations, often mimicking experimental device machine architectures. Plasma physics will always play a significant role in defining a fusion power plant but its concentrated focus in the design process will never lead to a successful power plant design. The Spherical Tokamak Advanced Reactor (STAR) fusion power plant design, introduced at the SOFT 2023 conference [1, 2], has evolved to determine if a compact spherical tokamak design based on ST physics principles could be defined where performance parameters might come within the range of the Gen III AP1000 power plant.

The STAR design incorporates features defined to provide the best chance of economic success and targets a compact device with concepts employed that permits the machine reactor core to be not much larger than the AP1000 reactor design. The STAR design incorporates 12 enlarged HTS TF coils that allow 24 large collapsible inboard-outboard DCLL blanket sectors to be removed through large vertical ports with minimal piping connections that emanate from below, not through the vertical ports needed for blanket access – an arrangement that improves the chance of meeting high availability operations. Design goals were set to develop a blanket design with high temperature capabilities to increase thermal efficiency and a physics defined liquid metal divertor with increased heat load capacity for a compact device, a divertor design that eventually will last the lifetime of the blankets [3]. Concepts that enhanced design simplicity were followed, leading to limiting the TF size to lower its cost through off-site construction, a simplified machine design with lower part count, employing an assembly scheme allowing the installation of individual components (TF, thermal insulation, VV) to lower assembly time, enhance component alignment and testing. Improved technology and physics opportunities are also being pursued to enhance the fusion power plant's economic viability – such as incorporating a CEA high efficiency neutral beam system and investigating spin-polarized fuel (SPF) to increase power output and reduce tritium requirements. SPF also could simplify the tritium fuel cycle and neutronics requirements, potentially lowering associated costs. In certain configurations, SPF might allow increases in shielding and HTS material to further increase power output.

Papers have been written [4, 5] questioning the ability to reduce the size of a magnetic fusion power plant due to performance, shielding requirements and structural limitations. Taken as presented, one might consider these arguments as accurate, but further insight and discussion can lead to a different conclusion which will be highlighted in this paper along with an update on the design progress.

A critical metric often not considered in the early stage of fusion’s development is economic viability. To be successful, fusion science experimentation, technology development and engineering must be cognizant of the economic environment in which fusion will operate to move the development process in a direction that can best deliver an economically viable, commercially competitive magnetic fusion design.

The Technology Roadmap for ARC, the First Fusion Power Plant

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Commonwealth Fusion Systems and its partners have advanced the construction and assembly of the SPARC tokamak, on track for operations in 2026 in Devens, Massachusetts. The initial objective of SPARC will be scientific demonstration of Q > 1 in a tokamak, with experiments then shifting to the goal of exploring operating regimes for ARC, the first fusion power plant. In addition to building SPARC, and as a participant in the DOE Milestone Program CFS has continued to advance the design and technology of the ARC power plant. ARC has been designed from the start to be as SPARC-like as possible to enable the translation of the SPARC subsystems (such as the magnets, heating system, etc.) to ARC with minor or no modifications. Due to innovative design choices, the few remaining subsystems that are specific to ARC (such as the FLiBe blanket and remote maintenance system) can be substantially de-risked in parallel to the construction of SPARC, enabling all ARC subsystems to be de-risked before construction and integration of the power plant. ARC has also been designed so that it is flexible enough to incorporate early SPARC learnings very late into ARC design and construction. This talk will present an overview of the full technology roadmap to get to ARC, showing a timeline of the resolution of notable critical-path scientific and technological issues such as the FLiBe blanket and remote maintenance system noted above, their dependencies, and the innovations and/or test facilities required to address them.

STEP INBOARD CONCEPT DESIGN ARCHITECTURE

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The role of the STEP Inboard System is to protect the central column magnets of the STEP Prototype Powerplant (SPP) from the heat, particle and neutronic loads of the plasma while also providing useful heat for power generation. The spherical tokamak geometry of STEP means the inboard radius drives the size, and hence overall cost, of the machine. The Inboard System is required to achieve its functions within a minimal radial space and be integrated with other systems of the SPP.

To address these challenges, a concept design for the Inboard System has been developed. An Inboard First Wall (IFW) subsystem manages the heat and particle loads of the plasma with long, slender plasma-facing components supported by a Structural Manifold. The IFW subsystem surrounds a High-Pressure Shield (HPS) subsystem that attenuates neutrons and manages the resultant volumetric heating with shielding blocks and embedded pipes that contain high-pressure coolant. The HPS subsystem surrounds a Low-Pressure Shield subsystem that further attenuates neutrons to levels that satisfy the central column magnet lifetime requirements – this is to be achieved with shielding blocks supported within low-pressure coolant channels of a vacuum-vessel structure.

Here, we will present the overall concept design for the Inboard System, the reasoning for its architectural division into the subsystems, the associated materials and design choices, the supporting multiphysics analyses, manufacturing assessments, the ongoing integration challenges, and the future STEP development activities for the Inboard System.

Electron Beam High Heat Flux Testing of Plasma Facing Components for Extreme Environment Operation

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Electron beam high heat flux testing of first wall and structural fusion materials allows for an understanding of the materials survivability and damage tolerance under exposure to high energy densities with tailorable heat flux raster patterns, cycling, and thermal loading rates. In-situ thermal analysis with an IR camera, TC integration, pyrometer measurements, and calorimetry through a chilled water loop allow for the thermal management/performance of the materials to be quantified during exposure to high heat flux irradiation. The high heat flux chamber can also be adapted to investigate the effect of plasma erosion utilizing an ion beam source and localized gas environment. Coupling the high energy density irradiation, cyclic thermal loading, and plasma erosion together generates an extreme environment exposure that next-generation fusion materials must endure. High heat flux electron beam irradiation also permits the ease of handling of materials post-testing for inspection and characterization. The High Heat Flux Facility (HHFF) at Penn State ARL can perform heat flux testing and thermal analysis to advance the engineering science of thermal management systems including hypersonics, plasma facing components, accelerator targets, high power electronics, high temperature heat exchangers for space applications, and validate computational tools used for design and performance predictions.

Additionally, analytical techniques (i.e. microscopy and X-ray diffraction) and a tensile load frame can be employed to quantify a resistance to high temperature and high energy density irradiation. The evolution of the materials microstructure, surface quality, chemistry, strength, toughness, and ductility will provide insight into potential resistance to embrittlement and swelling. In addition to experimental testing and evaluation, there is significant interest in utilizing thin film physical and chemical vapor deposition and bulk materials consolidation for the fabrication/synthesis of metallic, ceramic, and high entropy alloy material systems. Optimization of material chemistry and processing draws upon a thorough study of structure-process-property-performance relationships in an effort to engineer next-generation materials for fusion energy.

Building a high temperature liquid lithium flow loop: Lessons for liquid metal engineering

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Liquid lithium is being pursued by a multitude of fusion power developers worldwide as the tritium breeding material for commercial power plants. This design presents significant benefits compared with using lithium in other forms, but presents challenges due to its corrosive effects on structural materials.

To substantiate the designs of desirable liquid metal blankets for fusion power plants, the compatibility and performance of structural and functional materials with these systems must be understood under representative conditions. These conditions include, but are not limited to, variables such as temperature, impurity level, and flow rate. Parallel with understanding of liquid-sodium-cooled fast fission reactors, liquid lithium investigation for the fusion breeder blanket system must be conducted to a similar degree of quality and quantity, for substantiation of the breeder blanket system.

Oxford Sigma has been designing and assembling a high temperature, pumped, liquid lithium flow loop for liquid metal corrosion testing of fusion-grade candidate materials in blanket components, named LiFTOFF. The objective of this facility is for it to be used by the fusion community to generate quality data which is relevant to the engineering design, substantiation, and operation of future facilities and fusion power plants.

Throughout the phases of this project, many lessons have been learned in the process of designing and building a liquid lithium loop with the requirement of generating quality corrosion data. Materials and assembly testing has been undertaken for preliminary assessment of structural materials lithium compatibility, as well as for derisking of the facility design. Results from this have informed the loop design process and will be presented and discussed.

Issues encountered, solutions devised, and lessons learned will be shared for the benefit of the wider fusion community, to accelerate future developments in this area and to realise the commercialisation of fusion energy.

Current capabilities of the developed system and planned future upgrades will be shared, alongside data collected and analysed so far. In addition, opportunities for facility access and collaborative development will be discussed.

This programme is funded by the UK’s Fusion Industry Programme and builds upon an initial phase of corrosion testing in static liquid lithium conditions with Oxford University.

In-Situ repair of tungsten plasma first wall

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Blankets in a fusion reactor serve both the functionality of heat transfer for power generation and breeding of tritium. The blanket module is radially recessed from these limiters, and the armor needs to be thin to allow neutrons to transmit most of their power to the breeding blanket. This allows an efficient heat transfer from the armor to the first wall coolant. Effective limiters will thus eliminate the plasma fluxes altogether to the blanket armor, and only high energetic CX neutral fluxes, electromagnetic radiation, and neutron fluxes will impinge on the blanket armor. Blanket modules are generally large components requiring time consuming and costly downtime for their replacement. Although no actual design of a blankets exists for the various fusion devices, they have to be assembled and built in large segments (weighting in tons) for maintenance purpose. In-situ repair of the thin first wall armor is an attractive option to avoid this costly maintenance downtime. We expect erosion at the blanket armor to be ~2 mm at specific areas of the tokamak when maintenance needs to be performed.   In this project, an in-situ repair technique is being developed based on an AM module deployed on a remote-handling manipulator to repair the damaged armor surfaces.
We have developed a wire arc additive manufacturing procedure for the repair of 6 mm thick tungsten plates. We have successfully been able to fabricate a crack-free 4 bead deposition. Usually, tungsten is fabricated by the powder metallurgy route, as fabrication via fusion based processing routes requires exquisite process control and may lead to excessive cracking. The aim of the project is to prove the in-situ repair of tungsten armor in any orientation. We have developed a test-stand to represent multiple welding positions and will be demonstrated this year. After the deposition, we will also be developing methodology to improve the surface finish of the deposited surface that meets fusion requirements. An optical scanner is integrated into the design to ensure the quality of the deposited metal.

Significance of Heat Usage from In-Vessel Components and Primary Coolant Selection Criterion

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Fusion energy is an emerging clean energy source in the medium to long term. The United Kingdom Industrial Fusion Solutions Ltd is paving a pathway for a commercial magnetically confined fusion power plant, namely Spherical Tokamak for Energy Production (STEP), with the ambition of building a STEP Prototypic Powerplant (SPP) by 2040.
The thermal management of fusion reactors presents complex engineering challenges, particularly in managing the combined neutronic and radiative heat loads from plasma to in-vessel components (IVCs). These components experience spatially non-uniform heat flux distributions, constrained by coolant operating temperature windows and material limitations. The coolant selection process involves navigating multiple conflicting objectives such as high heat flux handling capabilities with lower pumping power, operating temperature ranges, recovery from loss of coolant scenarios, safety & handling implications, radiolysis under fusion neutron spectrum and the influence of the possible dissociation products in the structural material, neutron shielding/ transparency, and ease of containment. For instance, areas demanding high neutron shielding preferentially use water, while regions requiring neutron transparency and high temperature benefit from gaseous coolants like helium. Helium, while offering excellent neutron transparency for tritium breeding, substantially increases the primary coolant pumping power - approximately 150 MWe for EU-DEMO.
Additionally, large plant parasitic loads necessitate high power conversion efficiency, demanding a high operating temperature, whilst also integrating low temperature heat received from high surface heat flux plasma facing components. This heterogeneity in heat sources poses significant challenges for power cycle heat integration, particularly when aiming to maximize thermodynamic efficiency, thus pushing the boundary of conventional power cycles.
This paper addresses the above challenges emphasising the requirements of integrating multiple heat sources received from IVCs, delivered in different temperature ranges by linking this to the net power produced to demonstrate its significance whilst also outlining the minimum coolant temperature to realise the net power target of >100MWe for SPP. Furthermore, a holistic perspective of the methodology of down selecting the primary coolants is also presented in this paper, hence clearly outlining the necessity for having multiple cooling loops.

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Characterization of thin-walled hemispherical shells of reduced-activation ferritic/martensitic steel F82H using room-temperature press forming

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The reduced-activation ferritic/martensitic (RAFM) steel such as Japanese F82H is the leading structural material for fusion DEMO blanket. The plasma-facing surface of the blanket in the Japanese DEMO reactor has a thin-walled hemispherical shell structure, but its manufacturability and structural integrity have not been fully evaluated. As part of the reliability assessment, this study aims to assess the characteristics of hemispherical shells formed by press forming of F82H. In particular, we focus on the room temperature press forming process as a simple method that does not require heating. We also systematically investigate the material properties under different heat treatment conditions after processing to identify the optimal conditions for fusion DEMO structural applications. Since this structural part is made through plastic deformation, it will operate under extremely harsh conditions where irradiation-induced hardening and process-induced work hardening are superimposed under actual DEMO operating conditions. Additionally, the strength of F82H, which has been plastically formed into a hemispherical shell shape, is expected to vary unevenly from the top to the sides of the hemispherical structure. Thus, the effect of this variation in strength on the overall structural integrity must be considered. However, due to its thin-walled, curved structure, it is difficult to evaluate the strength required for structural analysis, even using standard test specimens. To address this, in addition to standard methods such as hardness and microstructure observation, we use the small specimen test technique to evaluate tensile fracture behavior. Furthermore, the authors have developed a methodology to assess fracture toughness using a local approach based on the Weibull stress concept and to evaluate ductility through combined analysis and experiments that account for stress triaxiality. Based on these achievements, this research further develops a small-specimen technique to apply it to thin-walled structures. Using this test method, the initial test results of evaluating the crack propagation behavior in the thin-walled hemispherical shell are also discussed.

Structural design and integration concept of an EC launcher for VNS

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The concept of a Volumetric Neutron Source (VNS) in a beam-driven Tokamak configuration for qualification of the tritium breeding blanket and other in-vessel components for a fusion power plant is currently under study. It shall be equipped with an Electron Cyclotron (EC) system with the aim to inject in a first stage up to 10 MW (for later upgrades up to 15 MW) microwave power into the plasma for core heating, impurity control and NTM stabilization at frequencies of 140, resp. 170 GHz. The current concept features the installation of an integral launching antenna in one of the equatorial ports of the VNS Tokamak with six steerable beamlines primarily for NTM stabilization and six open ended waveguides (OEWGs) for plasma core heating and impurity control. This paper presents the conceptual design of the EC launcher structure and the mechanical integration of the optical launching system. Besides the general design layout also the concept of an active cooling system, internal nuclear shielding elements, plant integration aspects and Remote Maintenance (RM) procedures are discussed.

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Status of the qualification of the CVD diamond disks for the ITER EC torus windows

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The torus windows of the ITER Electron Cyclotron (EC) Heating & Current Drive (H&CD) system enable the transmission of high-power microwave beams and serve as vacuum and safety boundaries, ensuring the confinement of hazardous materials such as tritium. ITER will be equipped with 56 windows, installed in the transmission lines on the torus side of the EC system. Each window consists of a polycrystalline chemical vapor deposition (CVD) diamond disk joined into a metallic housing. Being part of the first confinement system, the window and the diamond disk itself are classified as Protection Important Components (PICs). The qualification of the diamond disk must therefore follow specific procedures, defined and agreed in advance in a dedicated test plan.
At present, well over half of these ITER diamond disks have been characterized with respect to their dielectric loss, and further characterization is ongoing. The loss tangent parameter is of high importance as it determines the fraction of the microwave power absorbed in the disk during beam transmission. It shall be as much as possible low across the entire aperture area of the disk, so that maximum temperatures and thermal gradients are kept at the minimum, together with the resulting stresses. Accordingly, there are specific requirements that define the limits for the loss tangent. All the disks investigated so far meet these requirements and a summary of the results of the measurements, performed with dedicated Fabry-Perot resonators, is presented. Microscopy inspections and testing on polarization complete the investigations of the disks and are addressed as well.

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Predictive maintenance in fusion devices: application to condition monitoring of plasma-facing components

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With a view to reliability, availability, maintainability, and inspectability (RAMI) analysis for DEMO, achieving an availability of 30–60% while minimizing annual unscheduled shutdowns is a critical prerequisite [1]. Predictive maintenance (PdM) is a key strategy for ensuring regular, rapid, and reliable maintenance of the plant. By using historical sensor measurements, PdM allows estimating the remaining useful life (RUL) of components or subsystems, enabling proactive maintenance scheduling. Plasma-facing components, such as the divertor and first wall, are subjected to extreme thermal loads, intense thermal shocks, and bombardment by plasma ions, neutral particles, and energetic neutrons, making them highly vulnerable to damage. Real-time monitoring under these conditions remains challenging due to the limited number of available diagnostic options and the high computational demand of reactor-scale thermal response models. To address these challenges, this study focuses on infrared imaging data collected during steady-state heat load experiments on beryllium tiles until failure mechanisms, such as delamination or exceeding critical temperature threshold, occur. The experiments were carried out in Forschungszentrum Jülich, Germany, by means of electron beam test facilities JUDITH 1 and 2 [2]. This opens up the possibility of observing damage progression under controlled conditions. The present study explores multiple data-driven PdM approaches applied to infrared data and expert domain annotations, identifying patterns and anomalies indicative of damage. This includes shallow machine learning models applied to handcrafted image texture features such as gray level co-occurrence matrices, local binary patterns and Fourier transformed images, as well as deep learning models applied directly to infrared images. While offering distinct paths to damage prediction and varying levels of interpretability, both approaches demonstrate promising predictive capabilities and potential for generalization.

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46.412

Design and Evaluation of Gas Atomization System for Rapid Prototyping of Metal Alloy Powder Used in Additive Manufacturing of Fusion Reactor Components

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Wednesday Posters 2, Lobdell (Building W20 Room 208), June 25, 2025, 2:00 PM - 3:30 PM

Additive manufacturing (AM), particularly laser powder bed fusion (L-PBF), is revolutionizing the production of complex, high-precision components across industries. In nuclear fusion, where reactor materials must endure extreme temperatures and mechanical stress, AM methods are especially promising. However, current reliance on Niobium-based alloys, such as Glenn Research Copper 84 (GRCop-84), leads to long-lived radioactive waste due to neutron activation. To address this challenge, a novel small-scale gas atomization system was developed, enabling the rapid creation of custom alloys tailored for L-PBF. Unalloyed metal powders are resistively heated beyond their liquidus temperature and injected into a high-velocity Argon jet, producing fine alloy powders with optimized properties. The atomization process was refined through schlieren imaging and numerical analysis, driving continuous enhancement of nozzle performance. Resin AM enabled rapid prototyping and iterative testing of gas atomization nozzle designs. Additionally, a custom cold spray nozzle was engineered to accelerate and bond atomized powders onto sample coupons, which are subsequently subjected to ion and helium bombardment to simulate neutron damage. This integrated system streamlines the rapid prototyping and evaluation of reduced-activation alloys, paving the way for safer, more durable materials in the additive manufacturing of next-generation fusion reactor components.

Progress on the Spherical Tokamak Fusion Pilot Plant Pre-Conceptual Design

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Wednesday Parallel 3a - Next Steps III, Kresge Main Theater (Building W16, upstairs), June 25, 2025, 4:00 PM - 5:30 PM

This talk will present a progress update on Tokamak Energy's pre-conceptual design of a Fusion Pilot Plant (FPP), which was first introduced at the APS DPP meeting in October 2024. Tokamak Energy is one of the eight private fusion companies selected for U.S. Department of Energy's Milestone Based Fusion Development Program. A key early milestone on that program is the delivery of a pre-conceptual design report.

The Spherical Tokamak geometry enabled by High Temperature Superconducting (HTS) magnets offers some key inherent advantages over alternative approaches for fusion energy. In particular, the potential to access high confinement, high beta, and high bootstrap current fractions in a stable plasma configuration is an attractive proposition for economical power plants. We describe the physics and engineering design to harness this potential in a pilot plant.

This will be the main summary talk on Tokamak Energy's FPP design at SOFE.

Modularity and Iteration in the Levitated Dipole: Experiences from OpenStar Technologies

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[1]OpenStar Technologies

Wednesday Parallel 3a - Next Steps III, Kresge Main Theater (Building W16, upstairs), June 25, 2025, 4:00 PM - 5:30 PM

OpenStar Technologies is making rapid progress towards the Levitated Dipole Reactor concept for fusion energy. This rapid progress is a key feature of the dipole itself, which once certain engineering problems are solved, is reviled to be a highly modular MCF concept. OpenStar leverages this modularity to rapidly iterate, which creates a significantly faster pathway to mature the technologies necessary to see the concept derisked. In our first prototype Junior, modularity dramatically reduced the time, and therefore the cost, of multiple integration attempts, allowing engineers to develop and test systems for the real integrated environment. This capability was critical in the demonstration of the novel superconducting power supplies (flux pumps) used to drive the magnet current, where Junior's first plasma constituted the highest TRL demonstration of the technology to date. This tolerance to failure, and ability to retry and upgrade were particularly important in the absence of previous high TRL demonstrations. Essentially, rapid iterations allow teams to take and accept larger amounts of risk across the project timeline, without consequences being fatal to project goals.

Currently OpenStar is commissioning a second core magnet with modest improvements to Junior, now referred to as MK II and MK I respectively. This second magnet is a chance to improve on aspects of the MK I system, but more importantly formalise the iterative environment that has been so successful in maturing OpenStar's enabling technologies. This iterative program allows for a magnet to be in operation for plasma experiments at the same time as another is under upgrade or commissioning. This high availability of integration environment then allows technical teams to make progress at all levels of integration, almost simultaneously.

This iterative framework is then used to scaffold progress towards our next flagship device, Tahi, whose performance requirements are designed to define the functional limits and scaling rules of future dipole reactors. Tahi's aggressive engineering goals, and the technologies necessary to meet them, can be staged, implemented, and iterated using the Junior class system. This allows several of Tahi's risks and projects to be front loaded allowing critical lessons to be learned earlier, while adjustments are still possible. Overall, this presents a radical change in how MCF fusion systems can be conceptualised, staged, and demonstrated, and as an advantage of the dipole, justifies the ambition of goals we seek to achieve.

This talk will cover details of our previous iteration campaigns, our current plans for Mk I MK II iteration cycles, and some examples of how key technologies for Tahi will be demonstrated before Tahi itself if fully commissioned.

Conceptual Design of a Fusion Pilot Plant and its Role in Fusion Electricity Roadmap

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Wednesday Parallel 3a - Next Steps III, Kresge Main Theater (Building W16, upstairs), June 25, 2025, 4:00 PM - 5:30 PM

A four-stage approach to Indian DEMO was proposed recently where it was argued that a gross electricity producing compact pilot plant is absolutely essential to bridge the technology gap between present-day machines including ITER to a net electricity producing DEMO reactor [1,2]. Such a pilot plant will have to be smaller than the conventional DEMO and hence would naturally require high-Tc superconducting magnets for compaction, long pulse operation for the steady-state heat extraction, and energy storage systems for uninterrupted power delivery to the grid.

In this presentation, we outline the conceptual design of a compact fusion pilot plant of 3.6 m major radius, 4.2 T toroidal magnetic field, 2.4 aspect ratio with 3000 s fusion pulse duration and 1000 s dwell that is aimed to generate a fusion power of 300 MW with an electric gain of about 0.7. A solenoid assisted plasma startup with a flat top plasma current of about 10 MA with at least 60% of bootstrap is envisaged. Non-inductive current drive schemes using beams, RF and microwave are explored. The baseline scenario relies on H-mode operation with double null divertor and was derived using the PSCOPE module of the SARAS code [3]. The plasma simulations for such a case will be discussed.

We explore helium-cooled solid and liquid breeder blanket concepts for tritium breeding and the calculations show that the steady-state heat extraction from the blanket requires a pulse length of the order of an hour [4]. Due to the smaller size of the machine, breeding blanket is considered only on the outboard and the tritium breeding ratio will be less than unity.

The pilot plant aims to demonstrate the key technologies for burning plasma operation of the order of hours, tritium breeding, extraction and re-use, heat extraction and power conversion and the uninterrupted electricity delivery to the grid for a stipulated time. The detailed analysis of the long pulse operation, remote maintenance, HTS magnets and shielding will be presented along with the constraints arising from the uninterrupted operation for 75 days of the plant where the latter is a requirement prescribed by the grid code.

References:

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3. P.N. Maya and S.P. Deshpande, Fusion Science and Technology, 80:6, (2023) 741-765
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An Overview of Technology and Engineering Requirements for Realta Fusion's Next-Generation Axisymmetric Magnetic Mirror Fusion Devices: Anvil and Hammir

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Wednesday Parallel 3a - Next Steps III, Kresge Main Theater (Building W16, upstairs), June 25, 2025, 4:00 PM - 5:30 PM

Realta Fusion is advancing the axisymmetric magnetic mirror fusion energy concept working in collaboration with the University of Wisconsin-Madison. A brief overview and latest results of the currently operational Wisconsin HTS Axisymmetric Mirror (WHAM), a simple mirror using two, 17 T HTS magnets made by Commonwealth Fusion Systems will be provided. These experimental results in conjunction with RealTwin integrated plasma simulations motivate the construction and operation of a next-generation simple mirror called Anvil. Anvil will serve as a key remaining physics and technology risk-retirement platform operating at full commercial scale, with the primary physics deliverable to demonstrate the stable sustainment of "end-plug" plasma conditions needed for the construction of an axisymmetric tandem mirror fusion pilot plant.

Additionally, Anvil can be optimized to generate significant DT neutron wall loadings $\Gamma n \sim 0.4 – 1 +$ MW/m² with a large irradiation volume of up to ~ 5 m³, providing access to high DPA rates with a DT fusion neutron spectrum as a volumetric neutron source (VNS). Technology and engineering requirements for the realization of Anvil will be provided, including a path to optimize the device to serve as a VNS for a variety of use cases, such as the testing and qualification of fusion generator materials and test blanket modules.

Following the full-scale demonstration of end-plug plasma conditions in Anvil, two end-plugs will be connected to an extended central cell to construct Hammir, an axisymmetric tandem mirror fusion pilot plant (FPP). Technology and engineering requirements for Hammir will be provided and compared to those for Anvil. Hammir is being designed to exceed the requirements for a commercial FPP as defined in the National Academies of Sciences, Engineering, and Medicine (NASEM) report "Bringing Fusion to the U.S. Grid" by demonstrating an electric gain Qe > 1 while producing net-electricity Pe,out > 50 MWe for at least three hours continuously. An update on the current design point and targeted capabilities of Hammir will be provided.

MatDB4Fusion: A new initiative to collect, merge, and leverage material properties data

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Wednesday Parallel 3b - Materials: Neutron Sources, Testing Methods, and Databases, Sala de Puerto Rico (Building W20 Room 202), June 25, 2025, 4:00 PM - 5:30 PM

A drastic acceleration of efforts is currently perceived in fusion energy research due to two major driving elements: Major breakthrough results in plasma physics and enormous investments with a growing number of private companies that seek partnership. With progressing design iterations of different fusion devices and an increasing number of alternative approaches, more and more technical challenges arise by nature and there are some factors severely limiting the anticipated pace of solving them. One of the largest factors is the development of a set of basic materials suitable for various fusion operation conditions in harsh neutron irradiation spectra, qualified and validated in very expensive irradiation campaigns. Entities from several nations went through this process for three to four decades and accumulated data in large, but partial, databases of irradiated/non-irradiated material properties. However, these data are often not public and while there are data published in open literature, they lack important background information in many cases and the necessary efforts to collect them are huge. Additionally, the rapid development of machine learning (ML) capabilities in recent years gives hope for reduced design uncertainties, qualification programme durations, and improved lifetime predictions, when properly applied to suitable datasets with sufficient quality and statistically-significant quantity.

Due to this urgent demand for data support, an international working group emerged from SOFE 2023, led by Clean Air Task Force (CATF). With the aim to create a unique and well-maintained global database for fusion material properties that could benefit the whole community (public and private organizations) the “Material Database for Fusion” (MatDB4Fusion) was born. It will collect all kinds of material properties data relevant for fusion device design in a flexible but structured manner with quality control, supported by many entities from different countries. While it should primarily host data for open access to benefit the scientific community the most, it may also serve as a secure area for restricted data under intellectual property rights from different stakeholders to allow for secure scientific exchange and/or analysis (with approval of the data owner). Neutrality and internationality are considered as key-aspects and embodied by a steering committee as well as the cooperation with OECD-NEA as database host. MatDB4Fusion is aimed to act as a single reference for all fusion device design approaches to access a wealth of data that did not reach publication or attention, assess data gaps and reduce duplication of effort by improving focus of experimental campaigns, accelerate creation of codes and standards for qualification and provide sufficient data for machine learning solutions. In addition to the purpose and scope of the database project, the talk will focus on its current development state and ways to participate or support our efforts.

BB mock-up testing in IFMIF-DONES

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Wednesday Parallel 3b - Materials: Neutron Sources, Testing Methods, and Databases, Sala de Puerto Rico (Building W20 Room 202), June 25, 2025, 4:00 PM - 5:30 PM

Given the delay in ITER's schedule, the fusion community is looking for solutions to qualify the breeding blankets (BB) before their final testing in DEMO. Thus, ambitious and attractive proposals, like the Volumetric Neutron Source (VNS) [1], are being explored. In parallel, an experts' Working Group on Breeding Blanket and Fuel Cycle Development, analyzed other possible solutions. Among them, IFMIF-DONES was considered a suitable candidate since the reactions in its lithium target will produce an intense high-energy fusion-like neutron flux, allowing the development of different fusion-related experiments.

The main goal of IFMIF-DONES is to validate and qualify structural materials to be used in DEMO, within the so-called high flux test area. In addition, the medium flux area, with a larger irradiation volume, constitutes a perfect test bench for tritium technologies validation. It is important to note that the IFMIF-DONES engineering design has been developed to maximize flexibility, and at this stage, new experiments can be proposed. The present work analyzes the characteristics of the medium flux area, and introduces the concept of the Test Blanket Unit (TBU), a fraction of the BB considered representative of a whole BB segment. In other words, the TBU is conceived as a BB mock-up that can increase the TRL of this important component up to a level of 5-6 ("irradiation tests for design evaluation and improvements" [1]).

The main goals of the TBU are to demonstrate tritium breeding, tritium extraction, cooling performances… but also to perform multi-physics experiments in integrated testing, making IFMIF-DONES suitable for the BB qualification.

[1] G. Federici, "Testing needs for the development and qualification of a, breeding blanket for DEMO", 2023 Nucl. Fusion 63 125002. doi: 10.1088/1741-4326/ad00cb

A Data-Driven Approach for Evaluating Small Punch Test Results of Irradiated Materials Using Machine Learning Surrogate Modeling

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Wednesday Parallel 3b - Materials: Neutron Sources, Testing Methods, and Databases, Sala de Puerto Rico (Building W20 Room 202), June 25, 2025, 4:00 PM - 5:30 PM

The accurate determination of material properties is crucial for the design and assessment of fusion reactor plasma-facing components. Traditional methods for evaluating material properties often require extensive testing and large sample sizes, which can be unfeasible for studying irradiated materials or assessing in-service components. The Small Punch Test (SPT) has emerged as a valuable technique for characterising the mechanical properties of irradiated materials, including tensile strength, fracture toughness, and irradiation-induced hardening. However, challenges remain in interpreting SPT results and correlating them with macroscale tensile properties.

Current methods rely heavily on empirical correlations, which are often criticised for lack of accuracy and material dependency. Standards like ASTM E3205 and European EN 10371 link specific points on SPT load-displacement curves—such as maximum force, elastic-plastic transition, and plastic instability—with isolated tensile properties like yield stress and ultimate tensile strength. This approach limits SPT's applicability in engineering design, particularly in radiation scenarios where a comprehensive understanding of material hardening behaviour is critical. While inverse analysis can be used to determine material properties, it is computationally expensive and requires extensive modelling and analysis for each test.

This study introduces a novel approach to overcome these limitations by developing a surrogate model using Gaussian Process Regression (GPR) integrated with Finite Element (FE) simulations. This framework provides a computationally efficient, material-independent method for evaluating the effects of irradiation and temperature on material hardening behaviour, enabling the reconstruction of complete stress-strain curves from SPT load-displacement data. To achieve this, FE simulations were conducted on a wide range of hypothetical materials, utilizing combined Johnson-Cook and Voce-Law nonlinear hardening models tailored for irradiated conditions. The resulting data was used to train a GPR surrogate model, which effectively captures the complex, nonlinear relationships between SPT load-displacement data, irradiation-induced hardening, and material properties.

This method was validated using experimental SPT data from irradiated steels, demonstrating its ability to address the shortcomings of conventional empirical fitting methods, which often struggle with variations caused by chemical composition and irradiation-induced microstructural changes.

This advancement significantly improves upon traditional empirical models and computationally expensive inverse analysis and offers an accurate, material-independent, easy-to-apply, and computationally inexpensive framework for interpreting SPT results. This framework reduces the need for extensive material testing and modelling, particularly in high-temperature and high-fluence radiation environments, thereby saving time and resources while maintaining high levels of accuracy and reliability.

Introduction and Status of the Fusion Prototypic Neutron Source Risk Reduction Activity

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Wednesday Parallel 3b - Materials: Neutron Sources, Testing Methods, and Databases, Sala de Puerto Rico (Building W20 Room 202), June 25, 2025, 4:00 PM - 5:30 PM

In response to a U.S. Department of Energy (DOE), Office of Fusion Energy Sciences (FES) request for information in 2023, sixteen different concepts were submitted by the community for consideration as a fusion prototypic neutron source (FPNS). The proposed concepts vary greatly in approach, maturity, and the degree to which they accurately mimic a fusion energy system environment. To gain a better understanding of the proposed concepts, an FPNS risk reduction activity was initiated with representation from across the U.S. fusion community. The goal of the assembled team is to provide a consistent, objective, and unbiased approach to understanding and articulating the risks and benefits of different concept approaches to an FPNS. The assessment of each concept is broken into three topical areas: 1) the ability to mimic a fusion energy environment, 2) the ability to meet the performance requirements, and 3) the overall system maturity.

In this status update, the approaches to estimating system maturity, performance, and ability to mimic fusion conditions will be discussed. In the case of system maturity and performance, input from the concept proposers was critical to the effort which was then expanded upon by the FPNS risk reduction team. The effort to determine the ability of the source concepts to mimic a fusion energy environment is straightforward for some concepts and is complicated for others with neutron spectra that differ from a DT fusion neutron spectrum.

The risk/benefit analysis process begins with determining the neutron spectrum, gaseous, and solid transmutant production rates for each concept. That data will then be used to perform a thermodynamic analysis of the solid and gaseous impurity generation rates in four candidate material systems, namely Silicon-Carbon composites, reduced activation ferritic martensitic alloys, vanadium-based alloys, and tungsten. For each concept and each of the four materials, a multiscale materials modeling assessment of the neutron irradiation damage and gaseous plus solid transmutant accumulation and microstructural evolution will be performed. This will define tolerable limits for impurity generation from the perspective of changes to the thermal-mechanical properties and property degradation. Finally, an assessment of the impact of pulsed irradiation effects on damage accumulation and thermal-mechanical property changes will be performed to define limits on the pulse frequency, temperature, and pulse intensity.

The result of the FPNS Risk Reduction Activity will be a report to DOE FES in March 2025. Periodic communication of results prior to the study completion will be held with input from DOE and the concept proposers. This presentation will summarize the main conclusions from the report.

THE ITER VACUUM CRYOGENIC TEST FACILITY FOR CRYOPUMPS AND OTHER CLIENTS

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Wednesday Parallel 3c - Tritium, Fueling, Exhaust, and Vacuum Systems II, Kresge Little Theater (Building W16, downstairs), June 25, 2025, 4:00 PM - 5:30 PM

ITER extensively uses cryo-pumping for its large vacuum volumes such as the cryostat (~8500m3), the torus (~1330 m3), the neutral beam injectors (~180m3 each). Designed to meet ITER environment conditions such vacuum levels, radiation, magnetic fields and nuclear confinement, there are 8 cryopumps for Torus and Cryostat allowing a nominal pumping speed of 100 m3/s for Helium, with an 800mm all metal isolation valve allowing regeneration during plasma operation. These pumps are the second largest client of the cryoplant after the magnets (25% of the overall heat load) and they are the results of a 20 years development. The pumps have been delivered to ITER and the test facility will allow testing for the first time in ITER relevant cryogenic conditions and gas loads. This is a key opportunity before final installation to characterize pump performances and optimize the overall process control.

The facility is connected to the cryoplant for the supply of cryogenics to the cryopump. This is managed by a test cold valve box connected to the cryolines on one side and to the cryopumps on the other via flexible cryojumpers. The cryopump can achieve the desired conditions for pumping with a thermal shield loop fed with 80K gaseous helium and cryopanels fed with 4.5 K super critical helium. The Test-Cold Valve Box is based on the conceptual design of an actual torus cryopump cold valve box.

All operational modes of the cryopump and its systems will be tested such as cool down, warm up, normal and high temperature regenerations. The test tank where the cryopumps is inserted is based on the same regeneration volume of the final machine configuration (15m3) thus allowing full characterization of regenerations performances. Also the test cold valve box, just as the production ones, can provide cryogenic supply of an external client thus allowing testing and development opportunities for other ITER cryogenic systems.

The first phase configuration of the test facility has been completed end of 2024 and is now ready for first testing of the pre-production cryopump with inert gases injection. The production cryopumps will be then tested according with the installation plan and cryoplant availability. A second phase configuration is under development to add hydrogen/deuterium injection. .

This paper will describe the test facility main features, the test plan and the main results achieved from the first test campaigns.

The impact of parahydrogen on cryogenic hydrogen pellet production

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Wednesday Parallel 3c - Tritium, Fueling, Exhaust, and Vacuum Systems II, Kresge Little Theater (Building W16, downstairs), June 25, 2025, 4:00 PM - 5:30 PM

In fusion devices, cryogenic technologies have a crucial role, not only in superconductive magnet cooling but also in the formation of cryogenic pellets that are used for disruption mitigation, control and fuelling of the plasma of fusion reactors. At the HUN-REN Centre for Energy Research a support laboratory has been set up to study cryogenic pellet production, launch and shattering of large protium, deuterium, neon, and neon-hydrogen mixture pellets for the ITER Disruption Mitigation System (DMS).

A key element of these experiments is to optimise pellet creation time which is limited by the conduction of desublimation heat through the ice layer. It is known that hydrogen has two isomers, para and orthohydrogen and solid parahydrogen has significantly higher heat conduction at low temperatures than orthohydrogen occurring with 72% concentration at room temperature. Therefore, the catalytic conversion between the para and ortho energy states of the hydrogen can decrease the time needed for the pellet's desublimation. For studying this process a new cryostat with a para-hydrogen catalyst has been developed and tested.

In this contribution, we provide an in-depth analysis of the design and the preliminary results of using a catalyst. Pellet formation with and without a catalyst has been studied at different pressure and flow rates. The findings of this development will help improve the efficiency of cryogenic pellet production.

Development of a novel molecular cage material for the separation of hydrogen isotopes

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Wednesday Parallel 3c - Tritium, Fueling, Exhaust, and Vacuum Systems II, Kresge Little Theater (Building W16, downstairs), June 25, 2025, 4:00 PM - 5:30 PM

Energy efficient separation of tritium and deuterium gases is critical for the realisation of commercial nuclear fusion power. A common method of separation is cryogenic distillation. This requires liquifying the gases at ~20K using multi-stage distillation columns, which is an energy intensive process.

The University of Liverpool have developed a novel porous organic cage (POC) material that can separate hydrogen isotopes through kinetic quantum sieving. The university have successfully secured funding through the UKAEA's Fusion Industry Programme (FIP) to develop the technology and investigate its potential to improve tritium recycling and reduce energy consumption during Nuclear Fusion Power Fuel Cycle applications.

Through the FIP project, the University of Liverpool has improved the reproducibility of the production of the material, developing scalable, efficient production routes by exploiting flow chemistry. This has resulted in increased throughput for each synthetic step, as well as improved material properties of the final POC (e.g. increased crystallinity and smaller more uniform particle sizes). These properties have promise to improve the performance of hydrogen sorption and separations by allowing more efficient routes for the gas molecules to move through. Preliminary results have shown this new POC material to have increased hydrogen storage properties. The techniques being developed allow for enhanced control of materials properties, opening up opportunities to understand in more depth what makes these materials capable of challenging separations.

Trapping of combinations of protium, deuterium and tritium and partial separation has been demonstrated at the National Physical Laboratory (NPL) using a lab scale test rig. The test rig designed and built by NPL comprises of a liquid helium cooled cryostat, tritium process monitor and universal gas analyser. The nanoporous powder is loaded into a custom sample assembly that in turn is loaded into the cryostat. A carrier gas (helium or neon) is constantly flushed through the system. The cryostat cools the powder to 60K before the hydrogen isotope gases are injected. The gases are trapped by the powder and released during a temperature ramp. Separation is achieved by controlling the temperature ramp. The viability of trapping all three hydrogen isotopes has been demonstrated. Results will be presented that show the incremental improvement of protium-tritium separation with finer temperature control.

AtkinsRéalis have designed a new, upscaled test rig. The new rig offers improved test repeatability and the capability to explore a broader testing plan. Greater experimental control is offered over the existing rig by a bespoke cryogen-free cryostat for tighter temperature control and automated feed gas batching by mass flow controllers. A centralised control system enables operation and data acquisition via an HMI display, while an independent safety instrumented system and fully remote operation improves operational safety. With the project midway through the construction phase, the experimental performance of the rig will be reported on following initial commissioning and testing. Testing will focus on demonstrating material performance at higher flow rates, masses of POC material and tritium activities (up to 100 GBq) and target optimising the operating conditions for both separation performance and energy consumption.

Hydrogen Inventory Simulations for PFCs (HISP) in ITER

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Wednesday Parallel 3c - Tritium, Fueling, Exhaust, and Vacuum Systems II, Kresge Little Theater (Building W16, downstairs), June 25, 2025, 4:00 PM - 5:30 PM

Tritium inventory is an issue of critical study for the International Thermonuclear Experimental Reactor (ITER), which has an in-vessel safety limit of 700g of tritium. Simulating tritium transport through Plasma Facing Components (PFCs) is necessary to determine the tritium stored throughout the reactor. This research develops an open-source simulation tool, Hydrogen Inventory Simulations for PFCs (HISP), to simulate the evolution of hydrogen inventory in nuclear fusion reactors. HISP is being used to determine the optimal tritium removal strategy for ITER by simulating various cleaning scenarios and evaluating their impact on hydrogen inventory in ITER's PFCs. HISP breaks the problem into three parts: development of a binning structure to describe the geometry of a given reactor (in our case ITER), comprehension of output plasma data from plasma source codes (such as DINA, SOLPS, SOLEDGE, etc.), and set up of a requested scenario (such as one full power pulse followed by one cleaning pulse). HISP uses these frameworks to simulate hydrogen transport with Finite Elements Simulation of Tritium in Materials (FESTIM), producing hydrogen inventory estimations. In the case of ITER, the main chamber and divertor are split into 102 bins that are each simulated in one dimension. A total of 6 scenarios are being tested, which vary the duration, order, and scenario position of cleaning pulses. The considered removal techniques are Ion Cyclotron Wall Conditioning (ICWC), Glow Discharge Conditioning (GDC), tokamak plasma with Inner Strike Point sweeps, and baking. Future work includes the competition of the HISP simulation package for simulating complete ITER campaigns and determination of the most efficient tritium removal strategy in ITER.

The Design and Novel Fabrication of the Helicon Fueling Ring for the Material Plasma Exposure eXperiment

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Wednesday Posters 3, Lobdell (Building W20 Room 208), June 25, 2025, 4:00 PM - 5:30 PM

The Material Plasma Exposure eXperiment (MPEX) is a steady-state linear plasma device designed to expose neutron-irradiated materials to fusion divertor prototypic plasma conditions for the purpose of performing plasma–material interaction studies. The MPEX device will be capable of power fluxes of up to 40 MW/m2 and the ability to operate at steady state for up to 106 seconds with magnetic fields up to 2.5 T. The Helicon Fueling Ring is a water-cooled vacuum component responsible for fueling the plasma at the MPEX Helicon Source evenly and continuously. The design is comprised of a non-cooled stainless steel vacuum vessel with 6 water cooled “bricks” that shield the uncooled vessel. The bricks are tri-layer explosion bonded blocks of copper with TZM on the plasma facing side and stainless steel on the outside. The TZM layer protects the copper from the plasma. The copper layer contains pre-machined channels to improve heat removal and thermal performance. These channels are machined prior to the explosion bonding process. The stainless steel layer enables a welded interface for the cooling water tubing and gas line tubing on the outside of the brick. The bricks have been designed to deliver uniform gas flow at 6 locations around the plasma column via near-radial gas tubes. The fabrication of the vessel is conventional, but the fabrication of the bricks involves a tri-layer explosion bonding with pre-cut water-cooling channels in the middle layer. This novel approach may be applied to future plasma facing components that require water cooling in components made of dissimilar materials that are not commonly joined.

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Simplifications in 3D Neutronics Analysis:

Implications for Fusion Diagnostic Systems

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In the realm of neutronics analysis, accurate Monte Carlo (MC) simulations are essential for predicting system performance, optimizing design, and ensuring reliable operation. One of the most critical aspects of achieving high-fidelity simulations is the level of detail present in the CAD geometry used as the input for MC simulations. This presentation explores the impact of CAD model resolution on the accuracy of MC simulations, with a particular focus on Thomson Scattering Systems (TSS) diagnostic. This study investigates how varying levels of geometric detail, ranging from simplified to highly detailed models, affect the precision of simulated interactions, including particle flux, spectra, and dose distribution. The use of the MCNP code for these simulations will be highlighted, demonstrating how appropriate model complexity ensures the reliability of simulation outcomes, particularly in systems where high accuracy is required. The session will also address strategies for balancing computational efficiency with geometric fidelity, as well as practical considerations for model preparation. Finally, this presentation aims to provide insight into best practices for integrating CAD models into MC simulations, ultimately enhancing the accuracy and effectiveness of diagnostic systems integration in fusion reactors across a variety of applications.

Results from the Hydrogen Retention Mechanism Experimental Campaign in HIDRA

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Lithium (Li) as a plasma facing material is of great interest because of its ability to retain and pump recycled reactive atoms, decrease instabilities and increase plasma performance. The most recent Li experimental campaign on the HIDRA device at the University of Illinois, Urbana-Champaign, confirmed the previously observed evaporation of liquid Li from the HIDRA-MAT probe to the Helium (He) plasma edge causes the background recycling He to decrease and allows significantly higher performance of the plasma. The experiments showed that He was pumped when Li+1 ions appeared in the plasma and not when only Li neutrals were present. This most recent campaign, the Helium Retention Mechanism Experiment in a Stellarator (HeRMES) proved the location of the retention effect and hypothesized a possible mechanism for it. A wall heating element installed by the HIDRA wall was used to show that He is being retained on the walls as the Li is being deposited in defined stripes and cooled down at the HIDRA wall. Heating the wall element after Li evaporations into a He plasma resulted in desorption of He when the Li reached its melting temperature of 180 °C, confirming that He is being retained on the wall. A hypothesis for the retention mechanism is the physical co-deposition of He and Li on the walls of HIDRA, the He getting trapped by the Li as it cools down.

Our new experimental campaign aims to investigate a similar retention effect observed when Li is evaporated in-operando into a Hydrogen (H_2) plasma inside of HIDRA. The previously investigated Li behavior into a He plasma was important regarding the issue of He ash buildup in fusion reactors, but the investigation of the behavior between Li and H_2 is essential in giving us insight into how the H_2 isotopes, Deuterium and Tritium, might behave in the presence of Li. Investigating H_2 behavior in the presence of Li is a good first step for this. The results presented here will confirm the H_2 retention effect observed during the low-recycling regime in the presence of evaporated Li into a H_2 plasma. However, as opposed to seeing the retention effect happening when Li+1 ions emerge in the plasma, we expect to see H_2 retention with just the presence of Li excited neutrals. This is because H_2 will chemically react with Li and form compounds, like LiH, as opposed to He being an inert gas and not readily reacting with Li excited neutrals. Furthermore, the easier chemical reactivity between Li and H_2 might yield to higher required temperatures for the desorption of H_2 off the wall heating element by the HIDRA wall, and therefore leading to a different hypothesis for the retention mechanism as opposed to the He retention mechanism. This presentation will introduce the results from this new campaign and investigate the differences between the retention mechanisms of He and H_2 when Li is evaporated in-operando into the plasma.

Emission front control during strike-point sweeping at TCV in view of emergency control when re-attaching on DEMO

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In contrary to present-day experimental tokamaks or even ITER, EU-DEMO will have to operate with a minimum set of diagnostics in order to maximize the breeding blanket area and to cope with the harsh nuclear environment. Furthermore, strike-point sweeping is planned as emergency solution in the case of a re-attachment event as otherwise the divertor would get damaged in less than 2 seconds. For detachment control, spectroscopy is planned as main diagnostic with a few lines-of-sight (LOS) looking down from the mid-plane into the divertor with a strong toroidal component. The geometry of the LOS and the strike-point sweeping combine to make detachment control an open challenge.

In this contribution, we experimentally demonstrate emission front control during strike-point sweeping in TCV. The location of the emission front is calculated in real-time using the MANTIS camera images. Deuterium injection is used as actuator. Furthermore, a synthetic spectroscopic diagnostic (SSD) based on MANTIS is developed that integrates the emission along user defined LOS. The SSD measurements are used to reproduce spectroscopic measurements with a similar geometry as planned for EU-DEMO, i.e., looking down in the divertor from the mid-plane with a strong toroidal component. The SSD measurements are used to study the effect of strike-point sweeping on the measurements. We show that a control signal can be extracted from DEMO-like SSD measurements, even in the presence of a strong sweeping of the divertor leg. This is used to propose an initial approach to perform detachment control in EU-DEMO.

Design, manufacturing and experimental characterization of a first-of-a-kind ECH mirror equipped with TPMS

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In the Electron Cyclotron Heating (ECH) system of the Divertor Tokamak Test (DTT) [1], a series of mirrors are used to transmit and launch the microwave heating beams generated by a set of gyrotrons into the plasma [2]. These mirrors are subject to significant heat loads with peaks of several MW/m2 due to joule dissipation from the reflected radiation. Additionally, the launcher mirrors, located in regions where the magnetic field is non-negligible, are expected to experience eddy currents induced by the fast variation of the magnetic field during tokamak operation, in particular during plasma disruption [3]. As a consequence, being the mirrors in a high magnetic field, they are subject to high torques, the extent of which depends on the electrical resistivity of the mirror material. In the case of steerable mirrors, the induced torque should be minimized. An alternative option to highly electrically conductive Cu-alloys is thus required for these mirrors. Other materials, characterized by worse thermal properties, require most efficient cooling solutions to prevent deformations and stresses that could compromise the mirror functionality. AISI 316L has been chosen as a possible candidate material for the M2 mirror.

An emerging cooling technology, utilizing Triply Periodic Minimal Surfaces (TPMS), has recently been proposed to improve heat removal and enhance the overall performance of ECH mirrors [4]. TPMS are structured porous media generated by combining trigonometric functions, resulting in a high surface-to-volume ratio that is particularly advantageous for heat transfer in high-heat-flux components used in fusion machines [5]. Preliminary studies have been conducted to select the optimal TPMS topology, size, and porosity from various alternatives [4]. Based on these studies, a TPMS-based prototype mirror for the M2, made from AISI 316L, has been designed with a gyroid lattice having an increasing unit cell size moving outboard, surrounded by an annular manifold. The thickness of the reflective surface is 1 mm, while the minimum thickness of the lattice is 0.8 mm. The prototype is elliptical, with major radius 89 mm and minor radius 63 mm, and it has a curved surface for the reflective layer. It was manufactured using additive manufacturing and de-powdered by the private company Ellena S.p.A.. Thermal simulations predict that the prototype can withstand the target heat load, with peak larger than 2 MW/m2, maintaining a maximum temperature below 250°C and a pressure loss lower than 3 bar at a nominal water flow rate of 10 l/min. To validate these predictions, hydraulic tests are currently being conducted at Politecnico di Milano in a dedicated water loop, to assess the mirror hydraulic performance.

This work presents the results of the design, manufacturing, and initial testing of the M2 mirror prototype.

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Tokamak Divertor Optimization via Modular Construction Phase 2

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The DIII-D tokamak has undergone several Pumped Divertor installations and upgrades over the past three decades. Many of these programs initially envisioned successive installations of different divertors. However, due to a focus on core physics, each program was ultimately scaled back to a single semi-permanent installation. For over 20 years, the Upper Divertor used graphite armor tiles mounted on a toroidal ring of water-cooled plates which are stood off from the vacuum vessel wall to create a cryogenically pumped plenum volume. Unfortunately, this construction limited flexibility in divertor shape and required costly and time-consuming design, removal, and installation phases. This lack of flexibility gave rise to the Modular Divertor program.

In August 2023, the upper inner water-cooled plates were removed along with the accompanying resistance welded studs on the vessel wall. Phase 1, referred to as Shape & Volume Rise (SVR), was installed via new resistance welded studs, graphite armor tiles on the vessel wall, and one row of copper alloy pedestals with graphite armor tiles mounted on the pedestals. Installation was completed in March 2024. Unfortunately, the second phase of the divertor program had yet to be designed which eliminated the opportunity to design a stud pattern conducive to the first and second phase. The second phase of the Modular Divertor program was approved for design with the Physics Validation Review (PVR) taking place the week before the SVR installation was complete.

The methodology of the phased modular divertor program pays off in dividends after the first phase. The same core physics and engineering team was utilized to design the second phase. The time spent on establishing and iterating on requirements and design methods during the first phase was nearly eliminated. Starting design work immediately after the prior phase allows for lessons learned to be present in the minds of the designers. Physics Validation Review (PVR) closure to Final Design Review (FDR) went from 10 months to 6.5 months. This is more impressive considering the scope of Divertor 2 increased from 4 new tiles rows (SVR) to 8 new tiles rows, 1 new pedestal row (SVR) to 3 new pedestal rows, and a higher resolution of diagnostics.

Phase 2 is referred to as the Dissipation Focused Divertor (DFD). It’s often called the Chimney divertor due to the novel “chimney pump” located between the target and X-point. This is expected to stabilize the detachment front between the target and X-point. The primary goal of the phase 2 test campaign is to demonstrate a detached divertor target while maintaining a high-performance core. This chimney construction is substantially easier to implement with the modular divertor copper pedestal support structure. An increased emphasis was placed on baffling and heat dissipation relative to phase 1. Installation is planned for August 2025. The advantages and challenges of this phase are presented.

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Study of runaway electron beam terminations and wall loads in DTT

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Runaway electrons (REs) pose a significant challenge to tokamak operations, as they can potentially carry a substantial fraction of the pre-disruption plasma current. Accelerated to energies up to several tens of MeV by the strong toroidal electric fields arising after a thermal quench, these highly energetic electrons can locally damage the plasma-facing components by applying thermal loads reaching tens of MJ per square meter. This can result in significant melting and threatens the structural integrity and operational longevity of fusion devices.
This study investigates RE formation and the impact of RE beam terminations in the Divertor Tokamak Test (DTT), a facility currently under construction in Italy designed as a testbed for advanced magnetic and divertor configurations relevant to EU-DEMO. Using the non-linear magnetohydrodynamic code JOREK, coupled with the vacuum-field code STARWALL, we analyze realistic plasma scenarios of DTT. To reduce computational costs, we treat the thermal quench in a simplified way in 2D by increased transport coefficients and, instead, we focus on the analysis of the RE formation in the successive current quench phase and on the RE termination during the vertical motion of the beam.
Our findings indicate that during DTT's early operational phase (plasma current ~2 MA), the risk of RE formation is minimal when disruption mitigation systems adequately limit impurity injection. However, in full-current scenarios (plasma current ~5.5 MA) the potential for RE generation increases significantly due to its exponential dependence on the pre-disruption current. By assessing the resulting heat loads on plasma-facing components under these conditions, we aim to inform the design of effective disruption mitigation strategies, such as sacrificial limiters, to enhance the safety and reliability of DTT. This will also provide critical insights for the development of EU-DEMO, advancing progress towards reliable fusion energy technologies.

A Simple and Decoupled Power Stability Control Method for Gyrotron Long Pulse Operation

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Gyrotron is widely used in ECRH (electron cyclotron resonance heating) system for heating plasma in nuclear fusion experimental devices. Due to the radiation cooling effect of the gyrotron electron gun, the beam current gradually decreases during the long pulse operation of the gyrotron, resulting in unstable output power. This paper discusses a long-pulse gyrotron power control method, which controls the beam current by controlling the filament power supply power, so as to realize the control of the output power of the gyrotron. This method includes data acquisition, feedback control, and filament power control. These three parts are decoupled. The data acquisition is based on PXI. The feedback control is based on the PI control method. The filament power control is implemented over Ethernet using SCPI. This method can improve the stability of the output power of the gyrotron. This method can be conveniently utilized for ECRH systems that have been developed without gyrotron beam current feedback.

Thermomechanical Analysis of Tungsten-Copper Brazed Joints in the Divertor for Fusion Applications Using Digital Image Correlation

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The European DEMOnstration Fusion Power Plant (DEMO) represents a critical step toward achieving sustainable fusion energy. Positioned as the transitional phase between ITER and fully commercial fusion reactors, DEMO aims to demonstrate the feasibility of sustained net-positive electricity production. As fusion power continues to develop, the divertor—a plasma-facing component responsible for managing heat and particle flux and plasma confinement—plays a pivotal role in maintaining reactor performance and protecting internal components from extreme operational conditions.

Tungsten has emerged as a promising material for divertor armor due to its unique combination of properties, including high melting point, excellent thermal conductivity, and resistance to plasma erosion. When paired with copper alloys, which act as highly efficient heat sinks, the resulting tungsten-copper (W/Cu) joints provide a robust thermal management system capable of withstanding intense thermomechanical loads and irradiation damage. These joints are integral to ensuring the divertor's structural integrity under operational conditions that simulate heat fluxes of up to 20 MW/m^2. However, the thermomechanical behavior of these W/Cu joints during service life remains a critical area of study to ensure reliable performance in the extreme environment of fusion reactors.

This study focuses on the experimental analysis of the thermomechanical behavior of W/Cu joints, manufactured through vacuum brazing, under conditions replicating divertor mono-block conditions. Specifically, uniaxial high-temperature tests were performed on samples where a thin layer of copper was sandwiched between tungsten blocks. This configuration mimicked the thermal and mechanical stresses experienced during operation, including high heat fluxes and compressive loads. These conditions are known to induce creep-fatigue damage, a failure mechanism that becomes critical under cyclic thermomechanical loading.

To gain precise insights into the mechanical response of the joints, full-field strain measurements were obtained using the non-contact optical technique of Digital Image Correlation (DIC) up to 400 °C. This approach allowed for monitoring of strain evolution, and damage initiation and evolution within the joints generating new knowledge to predict the divertor performance in terms of structural integrity, damage accumulation, joint failure and identification of the required strain values for creep-fatigue service life assessment using design codes.

Neutronic Modeling of the Fusion Prototypic Neutron Source (FPNS) Design and its Optimization

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This work proposes a Fusion Prototypic Neutron Source (FPNS) to assess the neutron-induced transmutation products from the Deuterium-Lithium (D-Li) systems, which will be able to mimic the irradiating environment for a prototypic fusion power reactor. The FPNS design is composed of 10 concentrating deuteron beams with a total current of 40 mA in the energy range of 35 MeV to 60 MeV. The beams are designed to collide with a liquid lithium target with a variable thickness from 5 to 10 mm wrapped by stainless steel tubes. Then, it is associated with a sample container for the irradiating space at a volume of 150 cc using a helium gap with a variable thickness from 5 to 10 mm. The initially proposed design is accurately modeled using the well-known high-fidelity neutronic and transmutation analysis codes, PHITS and FISPACT-II, with the assistance of an in-house coupling computation platform. Several comprehensive series of sensitivity analyzes have been performed to investigate the impact of the deuteron beam (height and energy), the thickness of the helium gap between the lithium target and the sample container, the dimension and material of the reflector, and even the nuclear data library (deuteron library, neutron library, Damage library) on the ultimate figure of merits including damage dose rate (DPA), gas production. The optimized parameters are finally determined with the inclusion of neutronic performance regarding DPA, gas production per DPA, heating power rate, and radioactivity isotopic inventory, in addition to the comparative study of several sample materials including Eurofer97 (a European steel alloy), Silicon Carbide, Stainless Steel, Vanadium alloy, and tungsten.

Sensitivity Analyses of Tritium Production in ClLiF Molten Salt Liquid Breeding Blanket

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The ClLiF (LiCl + LiF) molten salt liquid breeding blanket [1] is analyzed in terms of the sensitivity of its tritium production rate (TPR) using Monte Carlo method. The sensitivity analyses is focused on the cross-sections of the materials present in the breeding blanket, neutron multiplier, and structural materials. Adjoint neutron flux, TPR, sensitivity profiles and integrated sensitivities of the TPR in the blanket to the cross-sections were calculated using OpenMC on a 2-D tokamak reactor, based on the radial build of the ARC-class reactor and ENDF/B-VIII nuclear data library. The calculated sensitivities show that the (n, p), (n, a), (n, np) reactions of 35Cl are the most important contributors to the uncertainties which were obtained using covariances from the TENDL-19 nuclear data library. The sensitivities of the TPR to the materials present in the reactor will be presented. Sensitivity analysis for another candidate molten salt breeding blanket material, FLiBe, will also be included.

Radiation transport model development with Gitronics applied to JT-60SA

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Gitronics is a novel, modular and Git-based approach for the management and development of radiation transport models being developed in Fusion for Energy. A pilot case study is undergoing for the development of an envelope-based MCNP model of the JT-60SA reactor. A private server hosted by Fusion for Energy is being used to host the pilot case. The benefits of the Gitronics methodology are evident and presented in this publication. The modularization of the inputs and separation of concerns have removed the challenges of working with a single monolithic input file as MCNP usually requires. The use of Git for version control and GitLab for enhanced collaboration and error tracking and project management is highlighted in this work. New features and systems to the reactor model are being added in a progressive way with the aid of Gitronics. These additions aim to improve the level of detail and accuracy of the model, and the assessments performed with it while being properly tracked and easily reviewed.

Multiphysics analysis of Li cooled divertor substrate during loss of coolant accident (LOCA)

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In the ongoing study of potential designs for Liquid Metal Plasma Facing Components (LM PFC), so-called "slow" and "fast" Li flow divertor concepts are under investigation. In these concepts, the LM flows on top of a solid substrate for heat removal and/or particle absorption from the plasma. A second coolant, either He or Li, flowing through the channels inside the substrate is used at the same time to mostly cool the substrate in the accident scenarios, such as the loss of surface coverage.
In the previous study [Y. Jiang et al., Fusion Sci. Technol., (2025)], the MHD/heat transfer effects of the Li flowing inside the substrate were comprehensively studied under the normal steady-state operation conditions. While in the present study, the multiphysics analysis is extended to the abnormal divertor scenario where the Li layer on top of the substrate does not provide a full coverage or even totally dries out for a certain period of time, so that the substrate becomes directly exposed to the incident high plasma heat flux. To address this situation, an integrated modeling is performed using an unsteady multiphysics model in COMSOL, coupling LM MHD, heat transfer, and solid mechanics, with the main goal of evaluating conditions under which the major material limits of the substrate can still be met.

Fabrication of Prototype Ion Extractor Grid and its 500 sec Long Pulse Experiment

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A Neutral Beam Injection (NBI) system works as an auxiliary heating system for Steady State Superconducting Tokamak-1 (SST) with a capacity to inject neutral hydrogen beam power of 1.7 MW at 55 keV. SST NBI consists of 3 multi-aperture ion extractor grids subjected to differential high voltages and each grid is made up of Oxygen Free Electronic (OFE) copper. The fabrication of ion extractor grids is complex due to the requirement of precision dimensional tolerances during the machining of the grid and other technology involved e.g. electro-deposition of copper for making embedded water cooling channels. This paper describes an in-house technology development for the fabrication of a prototype ion extraction grid of size 150 mm × 60 mm which consists of 4 wavy embedded semicircular (R1.1 mm) cooling channels. There are 19 number of through holes each of 8 mm in diameter lying between the cooling channels. To study the thermo-mechanical behavior in long pulse operation, this prototype grid is tested with a heat flux of 0.88 MW/m² for 500s pulse (2 shots with 30 minutes OFF in between the shots) generated from 9 kW Electron Beam (EB) power. Inlet water at ~26 °C is supplied to the water header with 0.08 kg/s flow rate at 4 bar pressure to remove the incident heat flux. The online surface temperature of the prototype ion extractor grid is measured with an Infrared (IR) camera. The average surface temperature measured is ~ 48 °C ± 2 °C. Inlet and outlet water temperature is measured using a Resistance Temperature Detector (RTD). The bulk temperature rise ΔTw is ~17.8 °C and 18.26 °C for shot 1 and shot 2 respectively, which corresponds to the absorbed heat flux of ~0.64 MW/m², and the absorption coefficient is ~37%. After the high heat flux test, the measured leak rate is ~3.9<10^(-10) mbar-l/s. This high heat flux experimental results are compared with CFD analysis and found good agreement. This study will help in the fabrication of an actual size grid.

Neutronics Analysis of the ITER Pellet Injection System

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Upon its completion, the ITER fusion device will be the world's largest tokamak, a toroidal machine designed to contain plasma through the use of strong magnets. As the largest fusion device ever to be constructed, ITER will present unique challenges in terms of modeling, manufacturing, and transportation, to name a few. The size of the ITER machine necessitates the use of large and complex neutronics models and sophisticated tools to simulate the behavior of radiation inside the machine and the impact to equipment and structures nearby.

Many systems are contained within the ITER structure, all of which contribute to the safe and effective operation of the machine. One such system is the pellet injection system, which is critical to plasma operation. The pellet injection system has three primary responsibilities: (1) to provide pellets for fueling of the plasma, (2) to provide pellets necessary for the mitigation of edge-localized mode (ELM) disruption events, and (3) to inject pellets of material intended to test the effects of impurities on the plasma. Port cells 4, 10, and 16 on the 1st basement level of the building around the machine, B1, house the primary equipment for this system.

The radiation condition of the port cell, which exists outside the vacuum and in the building surrounding the machine, is important to consider for many reasons. The most important among these reasons is the minimization of dose to workers in the access corridors around the pellet injection cask during maintenance periods. Additionally, there are many pieces of equipment inside of the cask which are sensitive to high-energy neutrons and whose positioning will depend upon what regions of the cask, or port cell in general, yield the lowest exposure. These needs, among many others, necessitate the calculation of the prompt radiation responses and responses during shutdown and maintenance periods inside of the port cell.

ORNL-TN, an expanded and improved version of MCNP5 v1.6, was used in conjunction with the variance reduction parameters generated by ADVANTG for calculations of prompt responses. To determine shutdown responses, a rigorous 2-step approach was utilized, coupling ORNL-TN with the nuclide inventory code ORIGEN and several auxiliary codes. The results of this analysis are discussed.

Putting the Sun in A Box – A Brief History of Plasma Facing Components

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Amentums' ongoing support to the development of fusion energy technology is grounded in over 40 years of experience . We continue to support the Joint European Torus (JET) at UKAEA; having served as ITER's construction management-as-agent contractor as part of MOMENTUM, we are now supporting UKAEA across their range of programmes from Robotic Handling, through to Design and Build projects, and the UKAEA STEP (Spherical Tokamak for Energy Production) programme.

We give an industry perspective on the development, design and manufacture of plasma facing components, having been involved in the design, development, and manufacture of PFCs from the mid-1990s. This presentation charts PFC design evolution from the laboratory to full scale prototypes of ITER First Wall Panels (FWPs), including:

· Early Beryllium tiled prototypes: Innovation of the tile bonding methodology

· Thermal Fatigue Mock-Up: Application of the established beryllium tile bonding on a water-cooled composite Stainless Steel – Copper alloy heatsink

· Semi-Scale Prototype: Expansion to a 1/6th section of an FWP

· Full Scale Prototype: First panel manufacture to meet the strict design and quality assurance requirements set by ITERs

· Advanced Design Mock-Up (ADMU): An investigation of design and process improvements and alternatives to improve in-service performance, reduce manufacturing costs of the ITER FWPs and improve manufacturability.

We examine the evolution of PFC design before progressing to the development and testing of manufacturing techniques used for the successful production and testing of these components. A focus is the Hot Isostatic Pressing (HIP) technology; used for the joining of dissimilar materials for both the heatsinks and plasma facing tiles. A key aspect is the understanding of the afore mentioned copper alloy, copper-chromium-zirconium (CuCrZr), heat treatment requirements to ensure overall heat sink performance.

To close, we discuss the transition of the technology to tungsten-covered components. This includes our design and manufacturing development for the Thermal Barrier Limiter (TBL) and Tile On Heat Sink (ToHS) components produced for UKAEA in support of their STEP Programme. We demonstrate how we have extended our manufacturing technology to tungsten components and outline the challenges to the achievement of durable and high performing tungsten tiled PFCs qualified for fusion power generation.

Innovative Approaches to the Material Plasma Exposure eXperiment (MPEX) Steady-State High Heat Flux Components Design, Fabrication, and Qualification

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The Material Plasma Exposure eXperiment (MPEX) is a steady-state linear plasma device designed to expose neutron-irradiated materials to fusion divertor prototypic plasma conditions to perform plasma–material interaction (PMI) studies. The MPEX device will be capable of ion fluxes of 10^23 m^−2s^−1, power fluxes of 10 MW/m^2, ion fluences up to 10^31 m^−2, and the ability to operate at a steady state for up to 106 seconds with magnetic fields up to 2.5 T. MPEX final design has been completed, civil construction work is complete, and procurement and installation activities are actively underway. MPEX has several high-heat flux (HHF) components, including the dump, water-cooled bellows, a limiter, skimmers, microwave absorbers, and the target. The peak heat flux intercepted by the dump is 1.3 MW/m^2 and the target is 10 MW/m^2. The limiter intercepts the plasma's last uninterrupted flux surface (with a calculated total power of 35 kW) to define the plasma size exiting the limiter and to prevent damage to the helicon window from back streaming. Microwave absorbers are used to minimize stray microwaves leaving the device's ECH heating section, which can inadvertently heat components that cannot be shielded or otherwise protected. The maximum heat load on the microwave absorber is predicted to be 241 kW. The two biggest engineering challenges for the HHF components were the effective heat transfer from the plasma-facing surfaces to the heat sinks and design for manufacturability (DFM). Test articles were produced for the dump, microwave absorber, target, and limiter to validate the manufacturing processes and to perform HHF testing for assessing thermal and mechanical performance and reliability. Innovative approaches employed to overcome the design and fabrication challenges of HHF components will be presented, including the selection of cooling geometry, material choices, and manufacturing processes for complex parts and assemblies.

Measurement of the vapor pressure of lithium alloys in the Lithium Alloy Vaccum Appliance (LAVA)

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Liquid metals as a plasma facing component (PFC) in fusion reactors have become an attractive alternative to solid PFCs. Liquid lithium is the top among them due to its low-Z nature, self-healing surface, and promotion of the low hydrogen recycling regime.
However, liquid lithium as a PFC can be hindered by its relatively high vapor pressure (about 7.5 mTorr at 800K). This introduces relatively high Z material into the bulk plasma, thereby decreasing the reactor's efficiency. However, alloying lithium may substantially lower the vapor pressure ($Pb_{83}Li_{17}$ expected to be about 1x10^-6 Torr at 800K). Initial alloys of interest are $Pb_{84}Li_{16}$ and $Sn_{80}Li_{20}$.

Measuring the vapor properties of these alloys is done in the Lithium Alloy Vacuum Appliance (LAVA) at UIUC. This experiment measures the vapor pressure of liquid metals using the Knudsen effusion technique supplemented by a quartz crystal microbalance and produces vapor depositions for elemental characterization. Due to the reactive nature of liquid lithium and its alloys, especially at elevated temperatures, the experiment is done under vacuum. LAVA has undergone various modifications and upgrades (dual turbo pump operation, background noise compensation, water cooling for increased sticking coefficients) in order to collect more reliable and accurate data. Currently, LAVA can achieve base pressures in the order of 1x10^-8 Torr and push the liquid metal samples up to 1200°C. Initial vapor compositions for $Pb_{84}Li_{16}$, as measured by Time of Flight Secondary Ion Mass Spectrometry (TOF SIMS), show 11 at. % to 100 at. % lead at temperatures from 300°C to 900°C, though the relationship is not monotonic. For $Sn_{80}Li_{20}$, TOF SIMS measured only lithium in the evaporate. To confirm that LAVA was working correctly, lithium vapor pressure was taken and found to be within one order of magnitude of the NIST values. Vapor pressures for $Pb_{84}Li_{16}$ were measured in the range of 10 mTorr to 100 mTorr for temperatures of 700°C to 900°C.

Overview of Helium Retention Results with Lithium

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One of the major issues facing fusion reactions is the fact that helium is a difficult atom to pump out fast enough where it will not become a core contaminant problem. In the early 2000's initial evidence that liquid lithium could retain and transport helium was shown. More recently experiments in HIDRA showed that lithium, is also able retain helium and provide a low recycling environment leading to marked improvement in plasma performance. The HeRMES campaign was able to show that the helium was indeed retained at the surface where lithium was being deposited and a co-deposition mechanism has been suggested as the likely method for doing this trapping. Furthermore, experiments at DIFFER and EAST using liquid lithium plasma facing components (PFC) in helium plasmas as shown showed similar results where the helium retention was observed. This paper will go over the HeRMES, DIFFER and EAST results seen with the helium retention and furthermore, also show that any inert gas will be pumped, since argon plasmas also saw the same retention mechanism in HIDRA.

Simulations of Plasma Facing Components with Tight-Fitting Twisted Tape Inserts under One-Sided Heat Flux Conditions for LIBs

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As advances in fusion energy and plasma containment continue, there is a greater need for research related to the supporting subsystems and component design. Liquid Immersion Blankets (LIBs) are a current fusion energy concept using FLiBe molten salt as the primary coolant. Within these systems, Plasma Facing Components (PFCs) such as the divertor monoblock experience significant impinging heat fluxes on the order of 1-10 MW/m2. These high heat fluxes are often exposed to one side of PFCs such as the divertor monoblock, creating an uneven heating profile. Adequately cooling PFCs, reducing thermal stresses, and remaining within material limits is an active design challenge in fusion energy systems. One way to increase the thermal performance of PFCs is through the use of passive heat transfer enhancements (HTEs). Adding HTEs such as twisted-tape inserts to the divertor or first wall coolant channels can lead to greater heat transfer capabilities. This increase in heat transfer is accompanied by an increase in frictional pressure losses, which need to be accounted for in finding optimal use-cases for HTEs.

This study aims to evaluate the thermal performance of FLiBe molten salt experiencing a high, one-sided heat flux within a tight-fitting twisted tape insert. Large Eddy Simulations (LES) will be done using the open-source Nek5000 computational fluid dynamics (CFD) code to analyze the heat transfer performance. Conjugate heat transfer will be accounted for between FLiBe and the tape insert, as well as between FLiBe and the divertor monoblock. A Reynolds number and Prandtl number characteristic of FLiBe flow in LIB systems will be used for the simulations. The goal of this work is to expand on previous work done by the authors to evaluate the performance of twisted tape-inserts for use in fusion energy systems. The thermal performance will be analyzed using the Nusselt number and Darcy friction factor. Averaged temperature profiles will also be analyzed to determine how the temperature profiles within the twisted tape are affected by the one-sided heating profile. This work aims to implement increasingly realistic boundary conditions to provide a range of applicability related to fusion energy systems.

Vacuum Brazing Process for Large Scale Grid Production in Negative Ion Beam Generation

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Large grids (~1 x 0.5 m²) used in negative ion beam generation are high-heat flux components that require active cooling. To meet this need, cooling channels are embedded within the grid segments. Traditionally, embedded cooling channels in these grids have been produced using electroforming, as seen in systems such as JET-PINI, BATMAN, ROBIN, and ITER-DNB sources. However, an effort has been made to manufacture these grids on a full scale using the cost-effective vacuum brazing method for ion source applications in non-neutron environments, such as the TWIN source (a two-driver RF-based negative ion source). Additionally, these economically manufactured plates with millimeter-sized embedded cooling channels can serve as heat exchanger plates in various industrial applications. The cooling geometry of the grids is optimized through finite element analysis, prototype development, and experimental validation [1]. This work offers insights into the development of brazing procedures, fixture design, and implementation for manufacturing large grid segments. It also outlines a comprehensive inspection and testing procedure for grid acceptance.

[1] Ravi Pandey, Mainak Bandyopadhyay, M.J. Singh, Jaydeep Joshi, Arun K. Chakraborty, “Design and analysis of TWIN source extraction system grids with indigenous manufacturing feasibility assessment”, Fusion Engineering and Design 155 (2020) 111552, https://doi.org/10.1016/j.fusengdes.2020.111552

Update on the ECRH system for ST40

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This year ST40 is installing a gyrotron to become an electron cyclotron resonance heating (ECRH) and current drive (CD) system currently on the ST40 spherical tokamak. It is a multi-frequency system operating at 104 GHz or 137GHz employing a gyrotron purchased from Kyoto Fusioneering capable of 1MW operation for a 2s pulse length.
The present status of the project will be discussed along with the commissioning and operational steps for operation on ST40. These will include the design of the ECR launcher which when installed and commissioned will enable the demonstration of conventional ECR heating and current drive using second harmonic X mode during the first phases of operation before moving on operations that include non-inductive start up, current ramp-up, and electron Bernstein waves (EBW) excitation and heating. The control system and its commissioning will also be discussed.

Interaction of a Hydrogen Plasma with Sn and Sn-Li Eutectic as a Potential fusion PFC Material

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A significant obstacle to the realization of a fusion power plant is the development of Plasma Facing Components (PFCs) that can withstand the extreme heat and particle flux incident on the first wall and divertor region. While solid PFCs struggle under the intense particle and energy flux from the plasma suffering damage such as microstructure growth, sputtering and melting, liquid metals have become a widespread potential replacement. Liquid metal's use as a hypothetical PFC has been gaining popularity due to its ability to self-repair damage as well as pump lost fuel and waste particles to create a low recycling edge. Currently, tin, lithium and lithium eutectics are the commonly considered liquid metals for use as a fusion PFC. It is therefore important to research the Plasma Material Interaction (PMI) between hydrogen plasma and the liquid metal PFC candidates. To that end the interaction between hydrogen plasma and a Sn as well as Sn-Li molten surface was studied at the Center for Plasma Material Interaction (CPMI). In both instances, Sn particles were observed being ejected from the molten surface attributed to the accumulation, and subsequent escape, of hydrogen bubbles underneath the molten surface. Where the ejection of tin shows a clear correlation with the hydrogen radical flux, increasing linearly with increased atomic hydrogen flux to the surface. With work still being conducted to fully understand the accumulation of hydrogen in and the ejection of material from molten Sn0.8-Li0.2 eutectic. However, the ejection of particles from both Sn and the Sn-Li eutectic makes them less appealing as potential fusion PFCs due to the likely hood that using them would cause PFC/wall material to enter and poison the fusion plasma.

Outlining Magnetic Topology in the SOL and Conceptual Non-Resonant Divertor Design for Eos Stellarator

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Thea Energy, Inc. will build a quasi-axisymmetric planar coil stellarator, called “Eos”, which will be used as a neutron source for tritium and medical isotope production. Conceptual plans for a non-resonant divertor system are proposed, based on research into the magnetic field topology and system power balance calculations.

We use several different code bases to compare results and baseline the calculations on our equilibrium. FIELDLINES (STELLOPT) and EXTENDER calculate the total magnetic field by applying the virtual casing calculation on the coil and plasma current contributions. FIELDLINES and DIV3D perform fieldline tracing to illustrate flux surface quality at the edge, particle trajectories beyond the last closed flux surface and exhaust locations via strikepoints on a notional vessel first wall, which is represented as a uniform offset from the plasma boundary. We have conducted a study to evaluate how fieldline topology changes with respect to plasma and design parameters. System power balance calculations include input power from neutral beams and ECRH heating, as well as radiative losses. We can estimate the median and worst case power magnitudes to apply along each fieldline, quantifying the heat flux using strikepoint density.

Areas of high peak heat fluxes of ~10s MW / m2 and average heat fluxes of 1’s MW / m2 are found in areas of high curvature of the Poincare cross section, aligning with the non-resonant divertor concept. Carefully designed divertor systems and materials are necessary to deal with expected heat and ion fluxes, including radiative cooling via gas puffing, actively cooled surfaces (first wall and target plates), plasma-compatible facing materials (i.e. with low erosion and impurity transport), pumping and neutral compression. Further work to design and characterize target plates and model PMI effects and edge physics (including transport and neutrals) are underway.

STEP Divertor – PFC layout , recent design changes and development activities

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The Divertor design for the STEP Prototype Plant (SPP) has undergone refinement during recent design iterations as part of the evolution of the whole plant design. Here we will provide an overview of the current design status of the SPP Divertor with a focus on recent design changes and ongoing technology development activities related to the Plasma Facing Components (PFCs).

The heat and particle loads seen by the SPP Divertors are extreme and vary spatially throughout. An overview of recent design assessments will be presented, which have defined the performance of multiple PFC options against criteria such as steady-state and transient heat flux handling, technological readiness and manufacturing considerations. Combined with an understanding of the spatially varying loads, these assessments have allowed efficient technology positioning throughout the divertor, providing a balance of performance and complexity.

The technology choice for the strike points is driven in part by the need to withstand Type-I ELMs, to address a key risk that ELM-free plasmas cannot be guaranteed through all stage of operation. Previous design iterations of the SPP Divertor considered Liquid Metal Armour (LMA) for this purpose. LMA PFCs show promise of being tolerant to extreme transient heat loads, but are complex to integrate with other systems. The current SPP Divertor design instead uses tungsten micro-brush PFCs at the strike point; a technology which offers reduced integration complexity compared to LMA, but also increased resilience to transient heat flux events compared to alternatives such as tungsten monoblocks.

Engineering design constraints inform the specification of key interfaces, such as the divertor wall shape and location of the vacuum pumping ducts. An overview of how these interfacing design constraints were achieved will also be presented. In addition to design work, an overview of recent PFC manufacturing development activities will be presented as well as future planned technology development and testing activities.

Novel Reduced-Activation High-Strength High-Conductivity Copper Alloys for Additive Manufacture of Fusion Reactor Components

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Additive manufacture (AM) of fusion reactor components enables materials and geometries unachievable through conventional machining. GRCop-84/42 Cu-Cr-Nb alloys used in Laser Powder Bed Fusion (L-PBF) of current LHCD launcher systems for the DIII-D tokamak are unsuitable for future fusion power reactors due to neutron-activation of niobium. We report development of a compact gas atomization system, and gas atomization of three new reduced activation copper alloys using TiCr2, TaCr2, or TaV2 Laves phase precipitates that maintain the favorable L-PBF properties of GRCop while eliminating niobium within the alloy.

A compact gas atomizer produced sample sizes of 0.05-0.1g of material per batch with a quick turnaround time enabling rapid low-cost alloy development and tuning of gas atomization (GA) parameters. A high temperature tube furnace alloys low-solubility elements at up to 2500°C, and an additive manufactured GA nozzle array generated a supersonic argon flow producing 10-100 µm powder diameters. Melting of elemental powders at 1800°C for 6min produced homogenous alloying for all three material combinations, measured by EDX. FIB sectioning showed precipitate formation in GA powders of all three material types, implying a suitable replacement for GRCop alloys. Numerical simulation of atomizer gas-dynamics of in-flight droplet cooling coupled to CALculation of PHAse Diagrams (CALPHAD) derived models of alloy specific-heat predict cooling rates on the order of ~1x105 K/s for ~30 µm diameter GRCop droplets. Modeling of cooling rate vs particle diameter and comparison to precipitate sizes in Focused Ion Beam (FIB) cross sections of GA powder vs powder diameter explore the dependence of precipitate-size on cooling rate that is extrapolated to precipitate refinement in L-PBF. Precipitates in these AMed alloys were similar in size and area fraction to copper alloys resistant to neutron damage, providing sinks for vacancies, interstitials, and helium generation that result in voids and material degradation. Similarly sized NbCr2 precipitates suppressed void formation under self-similar-ion irradiation up to 40 Displacements Per Atom (DPA).

A method of neutron damage simulation is proposed, where novel alloy powders are cold-sprayed onto adjacent strips on a target coupon and co-irradiated using self-similar-ions + helium in a DPA gradient along the length of each cold-sprayed strip. Powder particle diameter within each strip produces a varying powder-particle-diameter controlled precipitate size for each alloy. This technique will allow simultaneous testing of projected neutron damage effects in several candidate alloys and precipitate size distributions to qualify and optimize alloy composition for fusion reactor use, where high-strength high-conductivity components are needed in RF systems and high-heat-flux first wall and divertor areas. Work supported by US DOE under DE-SC0014264.

High-throughput assessment of primary radiation damage resistance in Nb-free Cu alloys for RF antennas

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In this study, we successfully established a correlation between in situ radiation evolution data obtained through transient grating spectroscopy (TGS) and the final microstructural damage in materials. The research focuses on different CuCrTa alloys manufactured via physical vapor deposition, which were subjected to self-ion irradiation with 7MeV copper ions up to 20 DPA. Simultaneously, TGS was employed to monitor the evolution of material properties during irradiation, for example the surface acoustic wave (SAW) frequency and thermal diffusivity. The SAW frequency is closely related to the material microstructure, being a property quite sensitive to the material atomic structure, where changes in vacancy concentration and the creation of dislocation forests can change this value. By performing transmission electron microscopy (TEM) on the irradiated samples, we quantified the damage in the microstructure post-irradiation by counting the number of voids formed. These voids serve as critical indicators of radiation damage, which significantly impacts material performance in extreme environments.
A key observation in our experiments was the instantaneous drop in the SAW frequency measured by TGS when the ion beam was turned on. This prompt frequency drop was found to correlate strongly with the final number of voids observed in the irradiated material. This relationship offers a powerful means of predicting radiation damage resistance within the first few minutes of irradiation, reducing the need for extensive irradiation times and post-irradiation characterization. By leveraging this rapid assessment technique, we were able to evaluate various compositions within the CuCrTa alloy system efficiently and identify the composition that exhibited better radiation damage resistance.
The implications of this work are significant for speeding up materials development for fusion applications, particularly in this work in the context of radio-frequency (RF) antennas for fusion reactors, however applicable to other parts of the reactor as well. The ability to assess radiation damage in near real-time represents a transformative advancement, enabling faster and more cost-effective materials screening. This methodology not only reduces the time and resources required for traditional damage evaluation but also provides valuable insights into the radiation response of materials during the irradiation process itself.
The approach developed here establishes a new pathway for material exploration, particularly for applications requiring high-performance alloys capable of withstanding severe radiation conditions. By down-selecting an optimal composition within the CuCrTa system, we have provided a strong candidate material Nb-free for RF antennas in fusion reactors, contributing to the broader goal of advancing fusion energy technologies.

Data-driven mode loss detection for the emitted radio-frequency beam of ITER gyrotrons

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Gyrotrons are millimeter-length coherent electromagnetic wave sources generally deployed in magnetic fusion confinement projects. In ITER's re-baseline, the Electron Cyclotron Heating system foresees employing an increasing number of gyrotrons to provide on-demand power for extended pulses. Their reliability relies upon maintaining the correct mode for a stable electromagnetic beam emission along the pulse. When a wrong mode is generated, the radio-frequency wave does not couple with the gyrotron output window, and the power remains inside the gyrotron body. The consequence is twofold: the plasma does not receive the expected power and the energy left inside the gyrotron leads to a potentially dangerous overheating.
To ensure correct gyrotron operations and prompt action in case of wrong mode generation, a dependable mode loss detection system is of crucial importance. Since radio-frequency measurements are not available in the current baseline due to the layout arrangement of the transmission lines, alternative detection methods must be explored. In this work, the effect of mode loss during operation on the acquired gyrotron data at different time scales was studied and the phenomenon characterized. In the millisecond time scale, the mode change is reflected in an electron beam current trend anomaly, namely an unexpected trend change not correlated to a variation of the system input power or the acceleration voltage. In a slower time scale of the order of few hundreds of milliseconds, the mode loss leads to an anomalous increase of the collector temperature. The collector is the component receiving most of the additional heat load caused by the wrong mode. Although some of the gyrotrons are equipped with thermocouples on the collector body, the detection of a wrong mode through calorimetry requires an extended amount of time, during which the overheating could lead to damages, or at least to a reduction of the gyrotron lifetime. In order to develop a fast mode loss detection, a data-driven method based on the electron beam current trend prediction during the pulse was developed and its performances were evaluated against a dedicated dataset.

Impact of the use of a dual laser system on the microstructure and mechanical properties of PBF-LB/M tungsten

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Tungsten is a desirable material for nuclear fusion due to its high melting point, strength and low sputtering rate. While it has typically been used as a plasma-facing component, there is growing interest in using tungsten for structural applications to expand the operational capabilities of future fusion reactors. The greatest limitation of using tungsten is that its brittle nature greatly hinders its manufacturability. There is interest in developing additive manufacturing techniques, such as laser powder-bed fusion (PBF-LB/M), to overcome the challenges posed by traditional manufacturing methods. Previous work has successfully produced PBF-LB/M tungsten samples but crack formation is commonly observed (the time dependence of which has been studied by Vrancken et al. [1]), which is not unique to additive manufacturing and is also observed in welding and joining techniques. Crack formation is usually mitigated by preheating the material above the ductile-to-brittle transition temperature (DBTT) of tungsten (200-400 °C) to minimise thermal shock. However, the work done by Muller et al. used preheat temperatures of up to 1000 °C and still observed crack formation [2].

While preheating above the DBTT does reduce crack formation in tungsten, cracking still occurs. This work looks to reduce the likelihood of cracking further by employing a dual laser system. The secondary laser beam is used to post-heat the sample, reducing the risk of thermal shock as the material cools. A range of secondary beam powers up to 500W and delay times up to 3 seconds between the beams are investigated. Samples will be characterised using a suite of microscopy techniques and mechanical testing in order to better understand the conditions leading to crack formation in LPBF tungsten components.

[1] Bey Vrancken, Rishi K. Ganeriwala, and Manyalibo J. Matthews. “Analysis of Laser-Induced Microcracking in Tungsten under Additive Manufacturing Conditions: Experiment and Simulation”. In: Acta Materialia 194 (Aug. 2020), pp. 464–472. ISSN: 13596454. DOI: 10.1016/j.actamat.2020.04.060. URL: https://linkinghub.elsevier.com/retrieve/pii/S1359645420303281 (visited on 12/31/2024).

[2] A. V. M¨uller et al. “Additive Manufacturing of Pure Tungsten by Means of Selective Laser Beam Melting with Substrate Preheating Temperatures up to 1000 C”. In: Nuclear Materials and Energy 19 (May 2019), pp. 184–188. ISSN: 23521791. DOI: 10.1016/j.nme.2019.02.034. URL: https://linkinghub.elsevier.com/retrieve/pii/S2352179118302382 (visited on 09/16/2024).

Estimation of the Coefficient of Friction in Dissimilar Metals of Cu to SS Friction Weld Joints for NBI Ion Extractor Grid Fabrication

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Neutral Beam Injection (NBI) is a well-established technique for heating and current drive in magnetically confined Tokamak plasma worldwide for fusion research. The Steady State Superconducting Tokamak (SST) is equipped with a positive ion-based NBI system that consists of 3 ion extractor grids subjected to high voltage and are placed in a high-vacuum environment. The ion extractor grids are made of copper and received a heat load of 1.75 MW/m² during beam operation. This heat is efficiently removed by water cooling channels embedded in the copper base plate. Inlet water is supplied at a pressure of 12 bars from a water header made up of SS304L stub pipes (30 mm long, 12 mm outer diameter, and 9 mm inner diameter), which are connected to the water cooling channels milled on a copper plate. These water header stub pipes are friction welded to the copper plate. A metallurgical study and detailed characterization of this friction-welded joint were essential to ensure its reliability in the operating conditions. This paper investigates friction welding between electrolytic tough pitch copper (ETP-Cu) and 304 stainless steel (SS) in a rod joint configuration with a 16 mm diameter, utilizing the continuous drive friction welding technique. This welding was conducted by varying friction pressure and upset pressure while other parameters remained constant. From the welding parameters, the "coefficient of friction" (COF) was estimated, which is found in the range of 0.18 to 0.40. The concept of the COF is widely applied in both science and engineering. The variation in COF during the friction welding process affected the torque, which ranged from 12.13 to 16.2 Nm, and the power, which varied between 2094 to 2902 watts. Changes in COF, torque, and power were governed by the applied pressure, which, in turn, influenced the strength of the welded joints. Notably, higher COF values corresponded to significant improvements in the tensile strength of the joint. The quality of welded joints was evaluated through various testing methods, including visual inspection, radiography using γ-rays, tensile testing, fracture surface analysis, scanning electron microscopy (SEM), and energy-dispersive X-ray spectroscopy (EDS). All samples fractured on the ETP-Cu side, away from the interface, with the maximum recorded tensile strength reaching 202 MPa. This work on the metallurgical properties and characterization of the friction-welded joint between stainless steel and copper is a key step toward in-house fabrication of ion extraction grids for Tokamak fusion research. In the future, the authors plan to perform an experiment of asymmetrical configuration, such as copper plate to stainless steel rod, and estimate the coefficient of friction and study the mechanical properties of each joint.

Shutdown Dose-Rate Calculations of Operational Facilities using OpenMC

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Our research group at the University of Tennessee-Knoxville (UTK) has been recently involved in studies that explore the use of the R2S methodology in OpenMC for dose-rate calculations [1-3]. Thanks to an emerging collaboration between UTK and scientists at the Culham Centre for Fusion Energy (CCFE), we have the opportunity to explore validations of an operating facility such as the Mega Ampere Spherical Tokamak Upgrade (MAST-U).

MAST-U is undergoing significant upgrades to its operational capabilities, with plans for regular physics campaigns through 2027 [4]. These upgrades will improve key plasma parameters, including an increase in plasma current from 1.3 MA to 2.0 MA, magnetic field strength from 0.52 T to 0.75 T, and total neutral beam injection (NBI) power from 3.8 MW to 5.0 MW. This study uses OpenMC, an open-source neutron and photon transport code [5], using the rigorous-2-step (R2S) workflow for shutdown dose-rate and heating calculations. The R2S approach, capable of comprehensive dose evaluations without prior knowledge of dominant nuclides, provides high-resolution data.

This study emphasizes the validation of neutronics simulations, comparing OpenMC results with experimental data from the MAST-U tokamak. Neutronics validation is crucial for ensuring the reliability of computational models in predicting fusion reactor behavior. By utilizing detailed CAD models and experimental datasets [6], the work aims to align computational predictions with experimental outcomes, thereby advancing the predictive accuracy of fusion engineering tools.

Building on previous research on material activation under fusion-relevant conditions, this study extends the analysis to shutdown dose-rate calculations, employing effective dose conversion coefficients from ICRP-116.

The overarching objective is to refine simulation methodologies by benchmarking against experimental data, enhancing their applicability to real-world fusion scenarios. Success in this effort will facilitate the development of optimized shielding designs and maintenance protocols for tokamaks, reducing radiological risks to personnel while improving operational efficiency [7]. This work represents a pivotal step toward integrating advanced simulation tools with empirical research, supporting the safe and efficient realization of future fusion energy systems.

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Analysis of Inductively Couple RF Power in the LUPIN ion source using electromagnetic simulation

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The Large, Uniform Plasma for Ionizing Neutrals (LUPIN) ion source is a test-scale inductively coupled positive ion source being studied as part of an upgrade plan for the neutral beam injection (NBI) system of the DIII-D tokamak. LUPIN has an RF coil antenna wound around the quartz tube plasma chamber with dimensions of 20 cm length and 10 cm radius. Up to 20 kW of radiofrequency (RF) power at 20 MHz is applied to LUPIN. It is operated at a chamber pressure of 0.6 Pa, to generate hydrogen plasma with density of ~ 1018 m-3.

To optimize the power coupling in LUPIN, a 3D electromagnetic model is developed using COMSOL Multiphysics. Analysis of the power coupling is crucial for maximizing coupling efficiency, minimizing energy losses, and increasing the total ion current delivered to the overall system performance. In the process, the model evaluates the system impedance of the LUPIN ion source. The model accounts for the skin effect in the RF coil and eddy currents, and the plasma is approximated using a complex permittivity. The simulation results provide recommendations to the experimentalists for impedance matching between the power supply and the matching network.

An internal Faraday shield protects the quartz chamber from the erosion caused by capacitive coupling and the resulting energetic particle fluxes by effectively minimizing interactions between the plasma and the chamber walls. However, this comes at the expense of some power losses. Therefore, this study also quantifies the impact of the internal Faraday shield on the power coupling efficiency and processes within the plasma. The electromagnetic fields within the source are analyzed both with and without the Faraday shield to evaluate its impact on the capacitive and inductive components of the power coupling, alongside other parameters influencing the RF matching network. 3D effects, especially the distortion of the fields by the Faraday shield slit geometry, could have a significant impact on the potential for erosion of the quartz and the power coupling to the plasma. The simulation results are validated against experimental measurements of the fields in LUPIN, ensuring reliability and accuracy.

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The DIONISOS Helicon Plasma Device - Capabilities and Modifications

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DIONISOS is a helicon steady-state linear plasma device at the Plasma Science and Fusion Center designed to study fusion relevant plasma facing materials. It can expose samples to both plasmas and ion beams simultaneously for analysis (ion beam analysis, IBA) or material damage, using a 1.7 MV Tandem accelerator equipped with a plasma source for light ions and a sputter source for heavy ions.

Currently, DIONISOS is being modified to accommodate materials wetted with liquid metals (and possibly molten salts). A wetting chamber with a manually operated liquid lithium dropper now connects to the main exposure chamber, allowing in-vacuo wetting and subsequent in-vacuo plasma exposure and ion irradiation of lithium-wetted materials. A custom-designed, magnetically coupled rotary bellows manipulator arm (with a 36-inch stroke) allows the sample holder to rotate and translate between the two chambers. The holder can also be actively biased and heated, enabling control of incident ion energy and sample temperatures up to 1200C.

Kinetic Modelling of Impurity Transport in the HIDRA Stellarator during Lithium Evaporation

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The extreme environment present inside future fusion reactors will require plasma-facing components (PFCs) that can handle high temperatures and damaging ions fluxes without failure or poisoning the plasma via sputtering of the PFC material. The use of liquid lithium as a PFC has drawn increased interest in recent years due to lithium’s low atomic number and the self-healing characteristics of flowing liquid metal divertors. Recent experimental campaigns on the Hybrid Illinois Device for Research Applications (HIDRA) as part of the domestic LMPFC program have identified and confirmed additional benefits of the use of lithium: helium retention behavior during in-operando evaporation of lithium into a helium plasma. Post-experimental observations revealed patterns characterized by ‘double streaks’ of lithium deposition on the vacuum-vessel wall. In this work, we introduce a kinetic GPU-accelerated particle tracer and compare these experimental observations with simulated impurity ion transport in the scrape-off layer on a full high-resolution 3D mesh of HIDRA’s magnetic field. The magnetic field is solved through direct application of the Biot-Savart law and corroborated against magnetic field measurements. Particle motion is modeled via the Boris-Buneman algorithm and parallelized on a GPU through CUDA. Results of our simulations display the same ‘double streak’ patterns as observed on HIDRA. Once validated, this code will be used in support of current and future experimental campaigns by predicting the impurity flux at proposed PFC installation locations.

Optimization of Capillary Porous System with Fast Flowing Liquid Lithium for Spherical Tokamak Advanced Reactor Design

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Liquid Lithium (LL) can improve performance of future fusion devices by creating a renewable surface on Plasma Facing Components (PFC) and simultaneously provide deuterium pumping to improve confinement. Capillary Porous System with Fast Flowing (CPSF) of LL introduced at PPPL [Khodak and Maingi, PoP, 2022] allows controlled LL delivery on the plasma facing surface and fast flowing LL under the porous zone enhancing heat and mass transfer. Numerical analysis using coupled model of the plasma edge and PFC analysis [Khodak, et al., NME, 2024] demonstrated the ability of the CPSF system to handle high plasma heat flux density at the divertor surface of NSTX-U while maintaining the acceptable PFC temperature. Plasma-PFC coupled model allows consistent assessment of PFC temperature distribution, which unlike heat flux density has a direct impact on the structural integrity   Application of CPSF in a future fusion power plant divertor conditions introduces additional challenges due to significant increase in total amount of energy which needs to be extracted. Several design modifications were introduced in CPSF to improve handling higher total heat flux: additional cooling of the substrate enhancing heat extraction; thermo-electric heat transfer enhancement in the porous zone; and multiple inlets and outlets of LL flow. This design enhancements were tested using Spherical Tokamak Advance Reactor (STAR) design developed at PPPL [Brown and Menard FED 2023]. Numerical analysis was performed using coupled numerical model which includes plasma edge code SOLPS-ITER and 3D Computational Fluid Dynamics (CFD) and heat transfer code CFX which was modified at PPPL to allow accurate and efficient analysis of the flows of liquid metals in strong magnetic field. Conjugate heat transfer allowed to simultaneous analysis in fluid, solid and porous domains Coupled model can handle strong interaction occurring between LL and plasma which includes evaporation, sputtering and energy exchange. This is especially important when CPSF is used as an evaporator for the lithium vapor shield divertor scenario. In present contribution we expand a coupled model to include the temperature dependence of the recycling coefficient at the plasma LL interface. Experimental studies [Morbey et al., NF 2024; Veleckis JNM 1979] show that retention of hydrogen species by lithium strongly depends on temperature of the plasma facing surface and decreases dramatically for the temperatures above 420°C. This dependency was included in as a boundary condition in both plasma-edge and CFD code together with temperature dependent evaporation and sputtering. Numerical results of CPSF LL divertor performance for various STAR operating scenarios will be presented.
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Universal Data Exchange Module of the IOC Layer in the ICRF Control System Based on EPICS

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This research aims to design and implement an ICRF control system based on EPICS. By using low-cost hardware such as Raspberry Pi to build the IOC layer and FPGA to construct a universal data exchange module, a distributed control system is established to achieve device control, data acquisition, and processing functions. The study will further explore the application of the EPICS system in real-time monitoring and data interaction, and optimize the system structure and performance using object-oriented design concepts, providing an efficient solution for ICRF control.

Verification of OpenMC Toolkit SPINS: Simulated Plasma Input for Neutron Source with MCNP on the MAST-U tokamak

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In this contribution, we present a comparison between OpenMC and MCNP neutronic simulations of the Mega Amp Spherical Tokamak – Upgrade (MAST-U) using a novel fixed source experimental plasma input toolkit, SPINS (Simulated Plasma Input for Neutron Source) developed within this work. To accurately predict neutron interaction in future fusion devices, a neutronic simulation requires detailed information about the plasma equilibrium, transport, and source geometry. Previous work from Segantin et al., 2024 has successfully benchmarked OpenMC against MCNP for fusion applications using a simplified source and geometry. Here we present a validation of the OpenMC SPINS toolkit with MCNP for fixed source problems utilizing a tokamak geometry and a plasma source. The MCNP calculations are performed utilizing an SDEF input from the JETTO plasma code and the plasma profile is input into OpenMC utilizing the SPINS library. The same 3D computer-aided design (CAD) geometry of the full MAST-U facility was used to create an equivalent constructed solid geometry (CSG) for the comparison. Compared to previous methods, SPINS integrates the plasma neutron source volume and samples neutron sources proportional to the predicted volumetric neutron production to minimize sampling error. This volume integration allows for the flexible input of simulated and experimental plasma profiles. The plasma neutron source accepts 1D or 2D ion density and temperature profiles and converts them to native OpenMC neutron sources. This process allows for the accurate recreation of MCNP sources within the OpenMC framework. The magnitude of the flux for various regions of MAST-U facility is directly compared and show the validity and robustness of OpenMC for fusion neutronic workflows.

ORNL fusion materials research: From next-generation development to scale-up for manufacturing and qualification for licensing

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The inability of currently available materials and components to withstand the harsh fusion nuclear environment requires development of new materials and an understanding of their response to this environment. Additionally, these new materials must have a means to scale up to industrial manufacturability in support of viable commercial fusion energy. Thus, the overarching goal of ORNL'sFusion Materials R&D effort is to provide the applied materials science and engineering support and materials understanding to underpin the ongoing Office of Science - FES program, in parallel with developing materials for fusion power systems. Within this effort, ORNL continues to be integrated with the broad U.S. and international fusion materials communities, fusion technology communities, and the burgeoning industry to commercialize fusion energy. This presentation overviews our current efforts and describes near-term plans.

Key ORNL assets for this fusion materials effort is the unique neutron irradiation capability of the High Flux Isotope Reactor (HFIR) and associated irradiation infrastructure include hot cells, needed for capsule/experiment disassembly and for testing highly radioactive specimens, and the Low Activation Materials Development and Analysis (LAMDA) Laboratory. MPEX (the Material Plasma Exposure eXperiment) brings new capability to couple plasma exposure with these neutron irradiation assets.

In all cases, fusion materials are being developed in a design-informed fashion where property improvements are led by fusion-relevant design studies and directed at advancing the material TRL. Our program continues to pursue development of low activation structural materials. With the largest effort directed at RAFM steels, advanced tailored reduced-activation (RA) steels, and SiC composites. Options for advanced steels include development of castable nanostructured alloys (CNAs), reduced-activation bainitic steels, ODS steels, and aluminum-containing ferrous alloys that promise improved liquid metal compatibility. Parallel to this is PFM development with an increased emphasis on understanding radiation effects, high heat flux testing and development of next-generation refractory alloys & composites. Recent efforts include the application of additive manufacturing technologies to the development of new tungsten PFMs and compositionally graded tungsten-RAFM transition systems. Exploratory investigations are on-going for future generations of additively nanostructured RA steels, AM SiC ceramics & composites, advanced vanadium alloys, heat-resistance Cu alloys, and ultra-high temperature ceramics. ORNL's fundamental modeling effort is directed especially at understanding experimentally observed behavior. Recent focus has been on the irradiation behaviors of the W-Re-Os system, exploring new options and high burn-up behaviors for solid ceramic tritium breeder candidate materials, and developing integrated PMI workflows complimenting expected MPEX needs.

ORNL continues to host collaborations with key international partners including the National Institute for Quantum and Radiological Science and Technology (QST, Japan), the Japanese National Institute for Fusion Science (NIFS)-university consortium, and the Karlsruhe Institute of Technology in Germany, acting for EUROfusion. A collaboration with the United Kingdom Atomic Energy Authority (UKAEA) has also recently restarted. While NIFS and UKAEA collaborations leverage primarily with the early applied fusion materials research, QST and EUROfusion partnerships aim to enable and qualify designs of DEMO blankets, exploiting ORNL's experiences of code-qualifying structural materials for nuclear fission reactors.

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DIII-D Electron Cyclotron Heating and Current Drive Systems, New Power Calibration Methods, and Expansion to 10 Gyrotrons

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The Electron Cyclotron Heating and Current Drive (ECH/ECCD) systems on DIII-D currently comprise five non-depressed and one depressed collector 110 GHz Microwave Power Products (MPP, formerly CPI) gyrotrons. The combined total injected power is 3.4 MW into the DIII-D plasma via six of eight installed launchers with steerable mirrors that are mounted in pairs above the mid-plane at 240° 255°, 270° and 285° in DIII-D port coordinates. A new method calibrates the generated power at the gyrotron using pulse lengths longer than one second with RTD-arrays installed along the waveguide to the dummy loads and cooling water calorimetry. Measurements using a 1 MW dummy load that was moved from near the gyrotron to near the launchers found transmission line losses consistent with previous such measurements. A seventh MPP gyrotron is expected to bring the total injected power to approximately 4 MW. EC expansion Stage-1 includes installing two new Thales TH1512 depressed collector 117.5 GHz gyrotrons (replacing one existing gyrotron) that will use existing power supplies, waveguides and launchers to establish eight systems with injected power of approximately 5.2 MWs. EC expansion Stage-2 is ongoing in parallel to install two additional ITER-type 1 MW gyrotrons, Kyoto Fusioneering model KF-GY003. These operate at 104, 137, and 170 GHz, though this variant will be tuned for optimal power at 104 and 137 GHz, closer to the optimum frequencies for DIII-D, rather than the 170 GHz required by ITER. Two new 60.3 mm inner diameter corrugated waveguide transmission lines and dual-launcher will inject power from below the mid-plane at 255° symmetrically with the existing launchers. Additionally, Stage-2 is procuring a solid-state high voltage power supply capable of independently running two gyrotrons. The EC expansion plans to bring the DIII-D ECH/ECCD capability to 10 systems with injected power approaching 7 MW for plasma experimental campaigns. We describe the latest status of the operational systems and power calibration methods, along with the plans and progress of the EC expansion projects.

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Thermostructural verification of the DTT Inboard First Wall Limiters

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The Divertor Tokamak Test (DTT) facility [1] is the superconducting tokamak under construction at ENEA C.R. Frascati. During operation, a set of undesired transient events (e.g., major disruptions, fast and slow vertical displacement events, fast discharges) could occur, causing significant electromagnetic (EM) loads to arise in the plasma-facing components (PFCs) due to currents induced by magnetic field variation. The Inboard First Wall (IFW) limiter of DTT will be largely affected. To protect the inboard vessel/in-vessel components during plasma transients, the current design of IFW limiters features a structure consisting of a CuCrZr coaxial tubes, armoured by tungsten monoblocks. Due to the low electrical resistivity of the cooling pipe, high electromagnetic loads are expected.
This work presents the results of the stress analysis performed on the components of the IFW limiter, with reference to the relevant loads (thermal and EM) that may occur in operating conditions. Finite-element simulations have been carried out in order to calculate the stress distribution in the limiter components; the calculated stresses have been compared to the permissible values according to the criteria established in the ITER SDC-IC. The results of the analyses can be used to assess the effectiveness of the current limiter engineering design in relation to the thermal and electromagnetic loads expected during the DTT operational phase.

[1] Francesco Romanelli et al 2024 Nucl. Fusion 64 112015

Testing and Characterization of Porous Tungsten for Liquid Metal Plasma Facing Components Fabricated by Cold Spray 3D Printing

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A major engineering problem in MHD fusion reactors is the counteracting of high heat and radiation fluxes that damage divertor and first-wall components. To combat these issues, Tungsten (W) has been selected as the material of choice for plasma-facing components (PFCs) due to its high melting temperature and thermal conductivity, as well as its low sputtering yield and tritium retention. Nonetheless, W PFCs routinely suffer severe thermal shock during pulsed operations. To increase the lifetime of PFCs, liquid metal dispersed through capillary porous systems (CPS) have been proposed as a solution since they can act as a heat exhaust and tritium recovery system. An important factor in the success of CPSs is their pore structure, as this affects the wettability and flowability throughout the material.
However, manufacturing of W is notoriously difficult due to its inherent hardness, brittleness, and high melting point. These characteristics limit the methods of working W, especially when making complex structures. Additive manufacturing (AM), or 3D printing, provides a solution to this issue, being a near-net shaping technology that overcomes traditional methods. Although high-density (> 99%) W samples have been made using AM, research on 3D printing porous W is still in the early stages. As such, this study seeks to make porous W using cold-spray AM. The W powder will be mixed with gas, accelerated past Mach 1 via a de Laval nozzle, and deposited onto a substrate. The step motor moving the substrate platform will control the hatch spacing and scanning speed. The effect of plasma-treated powders and different diffusion reactions due to placeholders on porosity will also be studied. The Liquid Metal In-Vacuo Injection (LIVIn) system, housed at the Penn State EMPIRES lab, will be used to measure the permeability of the porous samples.

Interfacial mechanical properties of multilayer structures for high heat flux components of divertor

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The high heat flux components of the divertor are typically designed with multi-layer metal composite structures, where the mechanical properties of the interface are critical to the success or failure of the design. To obtain the data of interfacial mechanical properties, the special test specimens and fixtures were designed for mechanical property tests of two interfaces in W-Cu-CuCrZr-316 L four-layer materials. The uniaxial tensile testing (UTT) and shear tests were performed to obtain the tensile and shear strength of the interface and its toughness at room temperature. The problems of difficult clamping in the interface tensile test and deviation of shear surface in the interface shear test were solved by the novel structural design of specimens and fixtures. The Cu-CuCrZr interface of the specimens were fabricated by two different processes: vacuum brazing and hot isostatic pressing (HIP), and the effects of these two methods on the mechanical properties of the interfaces were compared. The strength and fracture characteristics of W-Cu interface, Cu-CuCrZr interface and Cu layer were analyzed and compared by fracture morphology and fracture topography. The results show that the vacuum brazing seemed to be superior to the HIP.

Demonstration of Core Heating and Current Drive from the High-Power Helicon System at DIII-D

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Helicon waves, also known as fast waves in the lower hybrid frequency range, have long been predicted to be an efficient source of heating and current drive in reactor plasmas. The high-power helicon system at DIII-D uses a 'comb-line' traveling wave antenna sourced by a 1.2 MW klystron at 476 MHz to enable a first test of this technology. Recent upgrades have addressed issues with arcing in the transmission line, enabling coupled power levels of > 0.5 MW for pulse lengths of > 2 s. The system requires conditioning after each vent due to nonlinear dissipative processes such as multipactor-induced plasmas within antenna and vacuum transmission line components; the conditioning process was shown to be quick within 60 DIII-D discharges under constant plasma conditions and RF power with an exponential rise in RF on-time and with a corresponding increase in the fraction of the applied power reaching the antenna. The system was also shown to be load-resilient in ELMing H-mode plasmas with no indication of impurities associated with injection. Recent experiments have demonstrated core power deposition in the tokamak with a correlated local electron temperature response in both L- and H-mode plasmas. Changes in magnetic pitch were also observed with the Motional Stark Effect (MSE) diagnostic during helicon injection and on the resulting q-profile in L-mode plasmas, which indicates helicon waves are driving current in the core.

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Workflow Development for Design and Analysis of Divertor Components for Infinity One and Infinity Two Stellarators

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Development of an iterative divertor design workflow was undertaken in the course of assessing multiple stellarator configurations for use in the Infinity One and Infinity Two Fusion Pilot Plant. This resulting software system executes automated three-dimensional stellarator component generation for first wall and multiple divertor concepts, analysis of component manufacturability, application of feasibility constraints, and assessment of plasma facing component performance in multiple operational scenarios. This workflow was designed to address challenges with divertor design as it relates to complex three-dimensional shaping, plasma interaction, and manufacturing constraints. Special focus was paid to streamlining interfaces between stellarator component generation and analysis software. Initial component generation was then used to inform subsequent iterations. The flexibility and automation of this workflow allowed for feasibility assessment of divertor performance for multiple magnetic configurations. The tools developed can be used not only to design next generation divertors but also to plan operational scenarios in relation to divertor performance requirements. The components resulting from this workflow will be demonstrated in the Infinity One Risk Reduction Platform and results will be used to inform scaling to an Infinity Two Fusion Pilot Plant.

The Development of a High Temperature Testing Facility for Materials in Pure Liquid Lithium

Parker Z[1]
[1]Amentum

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Lithium is critical for breeding tritium to fuel fusion reactions. Pure lithium is therefore being considered for use as a breeder blanket due to its high breeding efficiency and the lack of generation of hazardous radionuclides. A significant challenge for the application of liquid lithium is material compatibility; many common structural materials are thought to suffer from severe lithium induced corrosion rates, particularly at high temperature, which could make building a commercial plant unfeasible. Currently, there are limited materials that are known to be suitable for use with liquid lithium due to a lack of research on corrosion testing of potential structural materials and much of the existing work has been focussed upon short time scales and the development of novel materials.
A bespoke testing facility has been developed with the capability to test a large number of materials simultaneously in liquid lithium at high temperatures (≤800 °C) for long testing periods (up to 2000 hours). A static testing rig with integrated cover gas monitoring allows for detection of off gassing; and the addition of controlled gaseous impurities with measurements on how much is absorbed into the liquid lithium. High purity lithium is provided for the corrosion tests by a dedicated purification system, also providing information on the efficiency of the expected purification systems for liquid lithium systems. The facility is supported by a wide range of analytical equipment which have been investigated to help develop methods to analyse lithium.
A 6-month experimental program was carried out to test the compatibility of candidate structural materials in liquid lithium. The study investigated the corrosion resistance of a range of potential structural materials such as ferritic-martensitic (Eurofer-97 and Grade-91) and stainless steels (316L, 316LN and 304L), refractory metals (tungsten) and other materials of interest (silicon carbide, vanadium 4-4, copper) including coatings.
The data generated has been collated to provide a baseline of data on the conditions under which corrosion occurs. The effect of long-term exposure to lithium of materials can be used to help develop materials qualification programs required for the design, build and operation of a commercial reactor.
Tests were conducted at relevant operational (~ 400-600 °C) and potential fault scenario temperatures (600 – 800 °C) for test durations between 24 – 2000 hours. Materials analysis was conducted to assess the resistance of materials. Techniques used include mass loss, dimensional analysis, optical microscopy, and scanning electron spectroscopy (including energy dispersive X-ray (EDX) and electron backscatter diffraction (EBSD)). High-quality microscopic images have been generated displaying the microstructural features of the samples. In this presentation we will present the results obtained to date and their significance to fusion power plants.

Research on Intelligent Identification Technology of Weld Defects in Fusion Reactor Vacuum Chamber Based on Deep Learning

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Wednesday Posters 3, Lobdell (Building W20 Room 208), June 25, 2025, 4:00 PM - 5:30 PM

The vacuum chamber of nuclear fusion reactor is composed of numerous welded components, and its structural integrity and durability are crucial to the performance and stability of device. Non-destructive testing (NDT) technology plays a vital role in ensuring welding quality and guaranteeing the safe and stable operation of the reactor. The precise and efficient identification of weld defects is an essential aspect of NDT, as the identification results directly relate to the reliability and safety of the device. In recent years, visual inspection methods based on deep learning, with their outstanding generalization and robustness, have gradually replaced traditional defect detection techniques, significantly improving detection efficiency and accuracy in industrial quality control processes. However, the vacuum chamber of nuclear fusion reactor has a very high density of welds, complex and diverse structural forms, as well as defects with complex and variable characteristics, which pose a huge challenge to the accurate classification of weld defects. In addition, the scarcity of non-frequent defect data samples and the imbalance of features can easily lead to overfitting of deep learning models, thereby seriously affecting the generalization ability of the models.
The core objective of this study is to achieve intelligent detection of weld defects in the vacuum chamber of a nuclear fusion reactor, thereby enhancing detection efficiency and accuracy. The research focuses on the following three aspects: (1) Constructing a comprehensive and high-quality intelligent detection dataset for weld defects in the vacuum chamber of a fusion reactor, providing a solid data foundation for subsequent model training and algorithm optimization; (2) Designing and building an intelligent weld defect detection model based on deep learning, which should have strong feature extraction and recognition capabilities to effectively deal with complex and variable defect characteristics; (3) In-depth research and optimization of the weld defect grading algorithm based on the intelligent detection model, to achieve accurate grading of defects and provide a scientific basis for subsequent maintenance and repair decisions. This study aims to contribute to the advancement and development of intelligent detection of weld defects in the vacuum chamber of nuclear fusion reactors.

Design of the next-generation MRS for Pacific Fusion, Z Machine, the NIF, and OMEGA

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Wednesday Posters 3, Lobdell (Building W20 Room 208), June 25, 2025, 4:00 PM - 5:30 PM

The next-generation Magnetic Recoil Spectrometer (MRSnext) is being designed to replace the current MRS for measurements of the neutron spectrum from an Inertial Confinement Fusion (ICF) implosion at the NIF and OMEGA lasers, and more recently, now at Pacific Fusion and the Z Machine. The MRSnext will provide far-superior performance and faster data turnaround than the current MRS, i.e., a 2× and 6× improvement in energy resolution on the NIF and OMEGA, respectively, and 20× improvement in data turnaround time. The substantially improved performance of the MRSnext is enabled by using electromagnets that provide a short focal plane (~10 cm) and unprecedented flexibility for a wide range of applications. In addition to being able to measure neutron yield (Yn), ion temperature (Ti), areal density (ρR), and plasma-flow velocity over a wide range of Yn, MRSnext will be able to directly, uniquely asses the alpha heating of the fuel ions through measurements of the alpha knock-on (AKN) tail in the neutron spectrum. The goal is to implement a radiation-hard electronic detection system capable of providing rapid data acquisition and analysis. The development of the MRSnext will also set the foundation for the more advanced, time-resolving MRSt and serve as a testbed for its implementation at the NIF.

FUSERO: JHR’s applicability to fusion research by neutron irradiation experiments, update of the project

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Wednesday Posters 3, Lobdell (Building W20 Room 208), June 25, 2025, 4:00 PM - 5:30 PM

The Jules Horowitz Reactor (JHR) is a new Material Testing Reactor (MTR) currently under construction at the CEA Cadarache research center in southern France, near the ITER site. It will serve (once in operation beginning of the next decade) as a major research infrastructure for scientific studies on material and fuel behavior under irradiation. For this purpose, it has been identified within various European roadmaps and forums (e.g., ESFRI, SNETP). The JHR is a pool-type reactor with a power output of up to 100 MWth. Its operation is expected to begin early in the next decade, functioning as a global user facility open to international collaboration.

The FUSERO project, a collaboration between UKAEA and CEA, aims to optimize the utilization of the JHR for fusion researchers. This presentation addresses the JHR's capability to support dedicated investigations of fusion-relevant materials and behavior under neutron irradiation. Specifically, the paper describes future experimental capabilities for ‘fusion’ material irradiations. The initial phase of the project focused on defining specifications for FUSERO test devices, based on the most recent EUROfusion call for tender for material irradiations, a 2023 facilities review, and a survey distributed to both the fusion and fission communities. In addition to straightforward “cook-and-look” irradiations, the FUSERO project emphasizes the development of test devices for characterisation and qualification irradiation programs, prioritizing fewer samples with advanced instrumentation.

Based on survey feedback and input from a diverse group of experts across fusion organizations and disciplines, the following key areas for development were identified:

A. Functional materials for diagnostic or current drive system windows (ceramic tests).

B. Cryogenic testing for magnet material irradiation.

C. Thermo-Mechanical Fatigue (TMF) testing for first-wall and divertor materials.

D. Metal-liquid and structure interactions.

To support these goals, we are developing a specialized thermomechanical calculation tool for feasibility and pre-design studies, alongside experimental studies to validate technological components for these innovative irradiation devices. This presentation will demonstrate the feasibility and updated status of each FUSERO device, including innovative features such as online loading and measurement under irradiation. These devices will complement several other irradiation tools developed to support materials research during the JHR's operational phase.

Thermal modelling of WEST tokamak considering 3D magnetic ripple effects

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Wednesday Posters 3, Lobdell (Building W20 Room 208), June 25, 2025, 4:00 PM - 5:30 PM

Magnetic ripple due to the finite number of toroidal field (TF) coils is a source of toroidal asymmetry within tokamak plasmas. This impacts especially the incidence angle of the magnetic field lines on the plasma-facing components (PFCs) which can increase for several degrees. This directly results in higher plasma heat flux to the first wall components and can significantly overload individual plasma-facing components. The numerical work presented here first describes the 3D magnetic field-line tracing (FLT) technique taking into account the full first wall of WEST tokamak. The code package written in C language by our team was used to achieve this task and was benchmarked against well-established PFCFlux field-line tracing package [1], analysing a segment of baffle tiles only. Then a new 3D field-line tracing simulation was performed on full WEST wall, taking into account detailed 3D description of the reactor vessel wall including the baffle, inner/lower divertor, inner/outer protection panels and antenna protection bumpers. The vessel surfaces were defined with a triangular mesh. Due to the complexity and size of the WEST first wall the KD-tree optimisation algorithm was implemented to speed up the search of field-line to triangle intersection. Parallelisation was also implemented using OpenMP and OpenMPI libraries to run the code on HPC infrastructure resulting in a significant decrease of the computational time. The 3D resulting surface temperature is then computed from a thermal model – based on OpenFOAM package [2] - able to use as input the 3D heat loads computed on large geometry and the temperature dependent material properties of WEST first wall components. For actively cooled components, heat is transferred to the coolant, defined with the temperature dependent heat transfer coefficient and the coolant temperature. For passively cooled components the temperature on parts protected from plasma exposure the temperature is fixed to a specific value. Thermal model gives an estimate of possible maximum plasma loads on the plasma facing-components. Its results can be further used for thermomechanical studies to predict component lifetime and help in the design of operational spaces for experiments. The results can also be used to predict hot spots that form on the wall or be used for synthetic diagnostic studies where radiation from the first wall can highly affect the interpretation of noise on diagnostic systems such as IR cameras, bolometer, etc.

[1] M. Firdaouss, et al. Heat flux depositions on the WEST divertor and first wall components, Fusion Engineering and Design, vol. 98-99, pp. 1294–1298, Oct. 2015.

[2] H. G. Weller, et al. A tensorial approach to computational continuum mechanics using object-oriented techniques, COMPUTERS IN PHYSICS, VOL. 12, NO. 6, NOV/DEC 1998

Materials development for the ARC fusion power plant

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Wednesday Posters 3, Lobdell (Building W20 Room 208), June 25, 2025, 4:00 PM - 5:30 PM

Commonwealth Fusion Systems (CFS) will deploy the first ARC fusion power plant in Virginia in the early 2030s. ARC consists of a compact tokamak featuring high temperature superconducting (HTS) magnets, a molten salt breeding blanket and primary coolant, and a replaceable integrated vacuum vessel. Delivering ARC will require rapid maturation of materials and processes for operation in a range of extreme environmental conditions including high temperatures, neutron, and heat fluxes in vacuum vessel components; deep cryogenic operation of magnet structures under cyclic loading and high fields; lifetime HTS performance under strain and shielded neutron flux; and a variety of FLiBe molten salt wetted surfaces. Some components, particularly those exposed to the highest neutron fluxes, are designed to be replaceable over the plant's operation to extend the overall asset lifetime. Others, such as the HTS and structures of superconducting magnets, must last the life of the plant. To address these challenges and others, CFS has established a materials and processing development program focusing on validating material performance in expected operational regimes where possible, conducting rapid emulation testing where required (ie. multi-beam ion irradiations in place of direct neutron exposure), and designing new material solutions where necessary. To enable early 2030s deployment, CFS requires deep integration with current and emerging supply chains to be able to provide industrial quantities of the power plant materials. The overall ARC materials program roadmap as well as the tool and capabilities required to meet this need will be described along with examples of success this playbook has enabled during SPARC deployment.

Direct Cavity Combiner for High Power Solid State RF Transmitter

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Wednesday Posters 3, Lobdell (Building W20 Room 208), June 25, 2025, 4:00 PM - 5:30 PM

Recent advances in fusion technology are accelerating the development and integration of commercial fusion devices. In order to maintain controlled fusion reactions, the plasma must be heated to great temperatures and efficient current drive must be established. The Ion Cyclotron Range of Frequencies (ICRF) have been demonstrated to be effective for these purposes, and for high magnetic field devices, 60-240 MHz radio frequency (RF) systems are envisioned.

Conventional vacuum electron device (VED) RF sources have a high life cycle cost with frequent maintenance, a large footprint, poor efficiency, and a fragile supply chain. Additionally, current solid-state solutions face similar footprint, costly combining and cooling obstacles that are resolved in DTI's solid-state approach.

Diversified Technologies, Inc. (DTI) is building a novel, patented, Direct Cavity Combiner (DCC) VHF Transmitter in a single high power, compact, and efficient amplifier under a Department of Energy Small Business Innovative Research (DOE SBIR) grant for use as a high power solid state RF transmitter. This transmitter, designed to scale to 1.5 MW, is built from multiple RF amplifier modules combined a single RF cavity, with high efficiency, low combining losses, and output power directly proportional to the number of RF modules feeding the cavity. This technology is an alternative to conventional megawatt-class VED RF sources, and overcomes the limited frequency range, reliability, and supply chain issues associated with tetrodes and similar VEDs. Because it eliminates the power combining losses and large footprint typical of conventional solid state amplifiers, it enables high power output and density. The basic transmitter technology can be readily tailored over a wide range of frequencies, which makes it applicable in several technologies, including high power microwave, high energy physics, fusion, radar, and broadcasting. DTI has demonstrated the viability of the DCC concept at L-band and UHF.

Under the Phase I effort, the full-scale 120 MHz cavity was partially populated with 31 modules demonstrating 42.7 kW of output power at 81% efficiency. In the next Phase of funding, DTI intends to combine 384 RF transistors, to achieve an RF output of 500 kW and to produce 100 kW of Continuous-Wave (CW) power. In this presentation, DTI will report on the latest design and test results of the VHF RF cavity and modules.

Update of the D1SUNED nuclear data package for the determination of decay photon related quantities in ITER

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Wednesday Posters 3, Lobdell (Building W20 Room 208), June 25, 2025, 4:00 PM - 5:30 PM

ITER will be the largest tokamak in the world and a key step in the path towards demonstrating the scientific and technological feasibility of fusion as a clean, safe, reliable and virtually limitless energy source. During its operation, ITER is expected to produce a total of up to $2.8 \cdot 10^{27}$ neutrons that will spread throughout the facility activating its infrastructure. As a result, intense decay photon fields will arise that may persist long after the machine shutdown. This is an aspect that needs to be considered in the design of the planned in-situ maintenance activities. The occupational radiation exposure (ORE) must be accounted for, and an ALARA program must be implemented to minimize it. In addition, decay photon fields will also have an impact on the In-Vessel Viewing System (IVVS) and the Divertor Remote Handling System (DRHS) operations as well as the cask transfer activities. A robust and reliable characterization and analysis of the decay photon fields is therefore necessary to design the ITER machine and tackle the negative consequences of radiation. That is the purpose of D1SUNED, which is the ITER reference computational code for the estimation of decay photon fields and their associated responses, like the shutdown dose rate (SDR). Common to many Direct 1-Step (D1S) based approaches, D1SUNED needs to be fed with a specific nuclear data set tailored for each problem and produced deliberately. These nuclear data are made by combining neutron transport and activation libraries. In 2018, the currently existing D1SUNED nuclear data package was produced. An activation analysis was performed to identify the radioisotopes and pathways of relevance for ex-vessel SDR analyses to be covered by the nuclear data package. However, since then, further activation analyses have been performed to better characterize the ITER radioactive inventory of concern for both ex-vessel and in-vessel analyses. As a result, radioisotopes and pathways have been identified that are not considered in the current D1SUNED nuclear data package. The need to include such radioisotopes and pathways, together with the need to upgrade the neutron transport, activation and decay data libraries on which the current D1SUNED libraries are based, has motivated the update of the D1SUNED nuclear data package. In this work, we present such update. The new nuclear data package has been prepared following the latest ITER and IAEA recommendations regarding the neutron (FENDL3.2b) and activation (TENDL2017) libraries. The package covers all the identified radioisotopes and pathways of relevance for ITER in-vessel and ex-vessel analysis. In addition, we also present the verification and validation (V&V) activities performed to guarantee the quality and correctness of the nuclear data package. These activities include the comparison against computational benchmarks and experiments implemented in the recent JADE nuclear data V&V tool. The new D1SUNED nuclear data package complements D1SUNED for the performance of robust and reliable ITER in-vessel and ex-vessel analyses according to the latest ITER and IAEA nuclear data standards.

ParaTAN: parametric neutronics modelling for tandem mirror type fusion devices

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Wednesday Posters 3, Lobdell (Building W20 Room 208), June 25, 2025, 4:00 PM - 5:30 PM

Magnetic mirror systems, once a prominent focus in fusion research, are regaining attention due to advancements in high field, high temperature superconducting (HTS) technologies and recent developments in plasma physics. These systems are attractive due to their simpler geometric configuration compared to alternatives such as tokamaks and stellarators. It is this simple geometric configuration that makes it easy to conduct fairly detailed parametric analyses of the first wall, breeding blanket and magnet shielding using the constructive solid geometry (CSG) tools available in OpenMC. Modern tandem mirror fusion design is enhanced and accelerated by flexible and efficient tools. The parametric modeling tool, ParaTAN, has been developed to support conceptual design and optimization of these systems. The modeling capabilities of ParaTAN encompass the first wall, blanket, vacuum vessel, various components of the central cell, and the magnets along with their corresponding shielding, all defined using a parametric approach. This flexibility enables efficient evaluation of tritium breeding blanket performance in terms of tritium breeding ratio (TBR) and heat loads, magnet shielding performance, and first wall and vacuum vessel damage and heat load, all of which are critical for guiding initial design decisions. To demonstrate its capabilities, a two-dimensional parametric study was conducted to investigate the tritium breeding performance of various breeder and multiplier materials. Shielding effectiveness of different materials was also evaluated, with particular focus on determining the minimum bore radius of HTS magnets in the end plugs of the tandem mirror device. Corresponding effects on the TBR, magnet heating and fast flux were analyzed using the open-source Monte Carlo particle transport code, OpenMC. The initial design space for magnetic mirror configurations is vast, with many possible features and tradeoffs to consider. ParaTAN accelerates exploration of this design space and supports faster decision-making regarding feature selection and optimization of tradeoffs, thereby streamlining the early-stage design process for tandem mirror systems.

What the SPARC Toroidal Field Model Coil Program should tell us about the future of fusion energy research in the US

Hartwig Z

Thursday Plenary and Awards - Andrew Holland, Fernanda Rimini, Zach Hartwig, Kresge Main Theater (Building W16, upstairs), June 26, 2025, 8:30 AM - 10:00 AM

Award Winner #2 Abstract

Fusion Technology Award - Tokamak Operations: from present experiments to next generation devices and Prototype Fusion Power Plants

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Thursday Plenary and Awards - Andrew Holland, Fernanda Rimini, Zach Hartwig, Kresge Main Theater (Building W16, upstairs), June 26, 2025, 8:30 AM - 10:00 AM

We are living in a very exciting time for Fusion Energy research.

There is an unprecendented, and justified, interest in our field: worldwide funding from public bodies and private investors is driving a multitude of fusion projects aiming at producing a viable prototype for a fusion power plant on ambitious timescales.

For those of us who have been involved in operating fusion experiments, and especially tokamaks, this progress generates a specific set of questions on how the next generation devices should be operated and how operators should be trained.

For most tokamak scientists, the term “operations” typically refers to running the tokamak and its subsystems for plasma experiments. However, in the context of preparing for the next generation, and further for a prototype power plant, we need to consider a wider scope than just plasma operations and include activities such as tritium handling, safety and investment protection, integrated commissioning preparation and, more generally, all the processes needed to ensure safe and effective exploitation of the devices. In addition, we believe that it’s beneficial to consider from early on in the design phase of a prototype Fusion Power Plant (FPP) how commissioning and operations will be organized, so that the Concept of Operations and the relevant processes can be rehearsed and adapted to the evolution of the device exploitation plan.

Looking ahead to the future becomes even more important when we consider preparing the personnel and processes for the prototype FPP. Statements are often made highlighting how the operation of FPPs will be more similar to a present Fission Reactor than a present tokamak experiment. The first FPPs, however, are likely to start operating very similarly to our present DT experiments, going from a plasma scenario validation and development up to full performance demonstration and, eventually, to reactor-like operations. Anticipating the requirements of this transition between present and FPP operation is one of the crucial activities for the fusion community over the next few years.

The challenge to prepare for safe operation of the next generation of devices and the first prototype FPPs will present an unmissable opportunity to build on the successes of the past years, from medium size, flexible tokamaks to the recent DT JET experiments. In addition, we must learn from best practices in industries with similar safety and investment protection requirements. Indeed, the involvement of industrial partners is essential and we need to be able to communicate clearly our requirements and be receptive to new approches.

Building an Industry: Commercial and Supply Chain Considerations for the Fusion Energy Industry

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Thursday Plenary and Awards - Andrew Holland, Fernanda Rimini, Zach Hartwig, Kresge Main Theater (Building W16, upstairs), June 26, 2025, 8:30 AM - 10:00 AM

The fusion energy industry is rapidly advancing toward commercialization, with companies planning to bring commercial fusion energy by the 2030s. This presents unprecedented opportunities and challenges for its supply chain. The FIA has found that supply chain spending by fusion companies have increased by near 25% over each of the last three years, with very significant increase predicted for the future.

However, scaling production of advanced components could present critical bottlenecks for scaling. This presentation will explore the industry's evolving commercial landscape, highlighting the need for long-term commitments, enhanced collaboration, and strategic investments to address supply constraints and workforce development. It will look into how government policy can build secure supply chains and durable partnerships. By fostering partnerships between fusion companies, suppliers, governments, and investors, the industry can build a resilient supply chain capable of supporting the deployment of commercial fusion power plants. Join us to discuss actionable solutions for overcoming these hurdles and accelerating the transition to a fusion-powered future.

The Critical Role of In-Vessel Assembly in ITER Construction: Scope, Challenges, and Strategic Approaches

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Thursday Parallel 1a - Design Integration, Construction, Assembly, Commissioning, and Lessons Learned II, Kresge Main Theater (Building W16, upstairs), June 26, 2025, 10:30 AM - 12:00 PM

The In-Vessel Assembly (IVA) is a cornerstone of the ITER Machine Assembly Program (MAP), responsible for integrating some of the most complex and technically demanding components into the tokamak. This includes diagnostic systems, heating and diagnostics port plugs, in-vessel coils, and key plasma-facing blanket and divertor structures. The successful execution of this scope is critical to achieving ITER’s mission.

The scope of IVA encompasses:

- Diagnostics, Instrumentation, Fuelling systems: Installation of diagnostic loops, sensors, fuelling systems, waveguides, instrumentations.
- Core Systems Installation: Assembly of in-vessel coils, blanket manifolds, blanket shield blocks and First Wall (FW) panels, and divertor cassettes.
- Port Plugs and Support structures: rails, diagnostics and heating systems port plugs and support structures.

This ambitious project faces several technical and contractual challenges:

- Integration Complexity: Coordinating interfaces across diverse systems while adhering to stringent tolerances and environmental constraints, with the as-built dimensions of the vacuum vessel and ports as yet unknown.
- Process Qualification: Developing and validating specialized welding and assembly techniques tailored to the unique requirements of ITER’s in-vessel components.
- Contractual Execution: Balancing detailed technical oversight with contractor capabilities, co-activity management, ensuring compliance with safety, quality, cost, and schedule commitments.
- Resource and Tooling Management: Procuring and deploying custom-built tools while managing contractor performance and ensuring readiness gates are met.

To address these challenges, IVA employs a multi-faceted strategy involving cross-program collaboration, robust contracting methodologies, and innovative process development. Leveraging a phased approach, the project integrates qualification milestones, optimization of sequencing and mitigation of technical and commercial risks.

This abstract highlights IVA’s pivotal role in the ITER construction timeline, underscoring its contributions to advancing fusion technology through timely planning and execution in one of the most intricate assembly projects in history.

Advances in the SPARC Project at Commonwealth Fusion Systems

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Thursday Parallel 1a - Design Integration, Construction, Assembly, Commissioning, and Lessons Learned II, Kresge Main Theater (Building W16, upstairs), June 26, 2025, 10:30 AM - 12:00 PM

Commonwealth Fusion Systems and its partners have advanced the construction and assembly of the SPARC tokamak (BT = 12.2T , R0 = 1.85m, a = 0.57m, designed to achieve Q ≈ 11), on track for operations in 2026 in Devens, Massachusetts. The initial objective of SPARC will be scientific demonstration of Q > 1 in a tokamak, with experiments then shifting to the goal of exploring operating regimes for ARC, the first fusion power plant. CFS is well into building and testing the superconducting toroidal field coils, the poloidal field coils, and the central solenoid for SPARC. Major supporting systems such as cryogenics, tritium handling, and ICRF heating are being installed and commissioned and will fully operate for a dry dress rehearsal (DDR) before SPARC operations. Engineering work is finalizing across most major SPARC subsystems, procurements are largely placed and hardware is arriving across the project. Pre-assembly work is underway with ongoing production for the Plasma Facing Components (PFC) tiles and Current Feeder System (FEED) components, as well as preparations and prototyping for next production lines starting soon. Tokamak assembly is ramping up with the arrival of major components such as the Cryostat base and Vacuum Vessel in 2025. The plant and plasma control systems based on the in-house neutrino framework are continually tested against hardware-in-the-loop (HITL) simulations in preparation for operation. This talk presents an overview of progress on SPARC.

Radiological and thermal management for large-scale tritium breeding experiments with molten salts

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Thursday Parallel 1a - Design Integration, Construction, Assembly, Commissioning, and Lessons Learned II, Kresge Main Theater (Building W16, upstairs), June 26, 2025, 10:30 AM - 12:00 PM

Tritium breeding blankets for DT fusion reactors require careful management of heat and radiation for facility and environmental safety, as well as tritium accountancy. Breeders with toxic properties, such as salts containing beryllium, add complexity to operational and administrative control plans. Through the Liquid Immersion Blanket: Robust Accountancy (LIBRA) project, we have undertaken the design and fabrication of tritium breeding experiments using 14.1 MeV neutron generators for irradiation of capsules filled with beryllium-containing salts at temperatures up to 700C. Tritium is collected from salt surface and wall permeation pathways, compared to neutronics simulations along with measured neutron fluences, and fitted using a simplified mass transport model. The goals of these experiments are to study fundamental properties of liquid breeder blankets, but also to project towards fusion reactor applications. Taking lessons learned from the smallest scales (0.1 to 1 L salt volume) and applying them to the design of larger experiments (100 to 500 L), where the neutron mean free path through the breeding volume is comparable to reactor blankets and tritium transport is dominated by the breeder medium, we present details of the radiative heating (experiment specific) and vacuum insulation (reactor relevant) of these experiments. Additionally, tritium permeation reduction with thermal spray alumina coating is considered to reduce the complexity of double-walled vessel design at these low levels of bred tritium, but considerations for scaling up to larger neutron fluences, and thus higher tritium concentrations, will be presented. Finally, we include measures taken to learn and adapt experiments taking into account beryllium hazard management when working with FLiBe breeding salt.

Progress in the development of the fusion breeding blanket technology for CFETR at ASIPP

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Thursday Parallel 1b - Blankets and Tritium Breeding III, Sala de Puerto Rico (Building W20 Room 202), June 26, 2025, 10:30 AM - 12:00 PM

In the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP), the water-cooled ceramic breeder (WCCB) blanket and the supercritical CO2 cOoled Lithium-Lead (COOL) blanket are being developed as blanket candidates for the Chinese Fusion Engineering Testing Reactor (CFETR). This contribution reports the overall design and R&D progress for WCCB and COOL blankets supported by the Comprehensive Research Facility for Fusion Technology (CRAFT) and the Burning Plasma Experimental Superconducting Tokamak (BEST) project.

The WCCB blanket features a mixed pebble bed of Li2TiO3 and Be12Ti as the tritium breeder and neutron multiplier, and pressurized water at 15.5 MPa as the coolant. Its feasibility is comprehensively assessed through analyses in neutronics, thermo-hydraulics, thermal-mechanics, tritium breeding, and nuclear safety. Related R&D activities are ongoing. The neutronics mockup experiment has been completed. Besides, the high heat flux testing facility, the water loop for thermo-hydraulic experiments, and the pebble bed testing facility have been constructed. The fabrication of blanket components is expected to be completed recently.

The COOL blanket is a typical dual coolant liquid blanket, utilizing PbLi at temperatures ranging from 460 to 600/700°C to cool breeding zones while employing supercritical CO2 at 350–400°C for cooling the first wall (FW) and structures. To address the challenges posed by high-temperature corrosive PbLi, electrically and thermally insulating SiCf/SiC composites are used as Flow Channel Inserts (FCIs), effectively mitigating the magnetohydrodynamic (MHD) effect. Extensive analyses covering neutronics, thermomechanics, thermal hydraulics, MHD, and safety have been conducted, confirming the feasibility of the blanket concept. In parallel, research and development activities are underway, including neutronics experiments on COOL mockups, construction of a high-temperature LiPb loop equipped with a 3T magnet, development of a high-temperature S-CO2 loop, and preparation of FCI materials.

To validate the breeding blanket technology, WCCB and COOL Test Blanket Modules (TBM) will be developed and installed in the middle ports of the BEST machine, a DT-operating fusion device scheduled for completion within the next few years. The design and testing plans for both TBMs will be briefly presented at the end of this contribution.

Operational Experience with Liquid Metal Circulation for the Century Repetitive Z-Pinch System First Wall

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Thursday Parallel 1b - Blankets and Tritium Breeding III, Sala de Puerto Rico (Building W20 Room 202), June 26, 2025, 10:30 AM - 12:00 PM

The central thesis of a shear-flow-stabilized (SFS) Z-pinch power plant [1] involves the use of liquid metal as an electrical conduction path, neutron-absorbing blanket, plasma-facing protective barrier, and thermal hydraulic fluid. Two generations of Forced Convection Loops circulating Bismuth (FCLBi-2 and FCLBi-3) have been fabricated and operated at temperatures up to 500°C to evaluate this concept's feasibility. These loops were integrated as subsystems of the larger Century fusion technology demonstration platform, which combines a repetitive pulsed power system with liquid metal-coated plasma electrodes and first wall. FCLBi-2 supported low-average-power SFS Z-pinch operations, while FCLBi-3 was designed to remove up to 100 kW of heat from the pinch vessel. This work discusses the design, fabrication, and performance of these systems, as well as plans for future operations. Details will be provided on the heat exchanger operation as well as pump performance.

[1] M. C. Thompson, B. Levitt, B. A. Nelson & U. Shumlak (2023) Engineering Paradigms for Sheared-Flow-Stabilized Z-Pinch Fusion Energy, Fusion Science and Technology, 79:8, 1051-1058.

Design Changes and Selection Methodology of the STEP Breeder Blanket

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Thursday Parallel 1b - Blankets and Tritium Breeding III, Sala de Puerto Rico (Building W20 Room 202), June 26, 2025, 10:30 AM - 12:00 PM

The role of breeder blanket in the STEP Prototype Powerplant (SPP) is to provide fuel self-sufficiency, net power, and the ability to provide high grade heat for industrial use. The design space for the STEP breeder blanket is constrained by the tokamak design choices of no inboard breeding, a double-null divertor architecture, the current low readiness of tritium breeding technologies, and the STEP operations targeting 2040. Additionally, the breeder blanket design needs be considered with the overall system performance and include requirements from safety considerations, tritium extraction, supply chain maturity, purge gas management and pumping power requirements.

The initial STEP concept proposed was a helium cooled liquid lithium breeder blanket with V-4Cr-4Ti as the structural material. This choice while advantageous from a tritium breeding ratio (TBR) standpoint had concerns surrounding safety and its impact on TBR, tritium extraction, and maturity of supply chain, as well as helium inventory management and pumping power requirements.

To mitigate these risks, the complete trade and design space of breeder blanket candidate materials was re-evaluated utilising wholistic metrics to identify alternatives that assure integration, delivery, and operational confidence while achieving good functional performance, simultaneously, on criteria of fuel self-sufficiency, high coolant outlet temperatures for power generation, and acceptable component lifetimes for high overall plant availability.

This paper covers the process that was followed along with a description of the methodology and metrics that led to the new STEP baseline breeder blanket concept. An extensive list of lithium-based breeder materials were considered and pared down based on comparative TBR simulations and qualitative engineering assessments. The shortlisted candidates were evaluated further on metrics of tritium transport and extraction, thermal management, structural material compatibility, safety, manufacturing, cost and commercial powerplant considerations. Evaluation work also included thermohydraulic analysis and breeder material coolant compatibility to give confidence in power generation performance.

Initial findings identified gas-cooled solid-breeder material blanket designs for the STEP. Further proof-of-concept design work addressed multiplier use optimisation, tritium permeation, breeder degradation & management, and structural material choice & activation.

The evaluation concluded with the selection of a STEP Breeder Blanket concept design consisting of a Carbon Dioxide cooled solid Li_2O breeder blanket utilising Titanium Beryllide multiplier with 15-15-Ti austenitic steel structural material and an oxide-based tritium permeation coating.

Recent progress of LIBRA project and new TBR measurements

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Thursday Parallel 1b - Blankets and Tritium Breeding III, Sala de Puerto Rico (Building W20 Room 202), June 26, 2025, 10:30 AM - 12:00 PM

The LIBRA project aims to develop the liquid immersion blanket concept (LIB) as a scalable solution to achieve tritium self-sufficiency in fusion power plants [1]. Tritium breeding is critical for sustaining fusion reactions, relying on the interaction of lithium-containing materials with high-energy neutrons to produce tritium. The LIB concept uses molten salts as both a tritium breeder and a heat transfer medium, offering a promising pathway to meet the tritium production demands of future fusion reactors.

The BABY experiment, a central component of the LIBRA project, was initially designed to investigate tritium breeding and transport at a 100 mL scale using molten salt (ClLiF) irradiated with 14 MeV neutrons from a deuterium-tritium generator. Although the 100 ml setup provided valuable insight, it was limited by scale and only measured tritium released from the salt’s free surface [2]. To address these limitations, the BABY experiment has been upgraded to a 1 L scale, allowing more representative testing and the addition of an outer vessel to capture tritium permeating through the crucible walls.

Initial experimental campaigns with the upgraded setup demonstrated excellent reproducibility, with a Tritium Breeding Ratio (TBR) of approximately $2 \times 10{-}3$, a six-fold increase compared to the TBR of the 100 ml scale experiment of $3.57 \times 10{-}4$ - thus establishing a new record for TBR. These results align closely with OpenMC neutronics simulations, validating the experimental and modelling approaches.

Interestingly, no permeated tritium was detected in the outer vessel, suggesting mechanisms such as preferential surface transport, oxide layer formation, or secondary permeation losses. These findings warrant further investigation to fully understand the dynamics of tritium transport and retention.

Future experiments will explore alternative breeders, including FLiBe, perform redox control studies, and evaluate the impact of varying purge gas compositions. Insights gained from these studies will guide the design of LIBRA-Pi, the next-generation experimental platform, bringing the LIB concept closer to achieving tritium self-sufficiency in fusion power plants.

Four years of CRANE: Developing computational physics research skills through community-centered education and mentorship

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Thursday Parallel 1c - Setting Up Fusion: Supply Chain & Workforce Development, Kresge Little Theater (Building W16, downstairs), June 26, 2025, 10:30 AM - 12:00 PM

The Computational Research Access Network (CRANE) is a free virtual workforce development program designed to foster excitement and community for undergraduate, graduate, and non-traditional students considering careers in plasma sciences. CRANE teaches Python-based research skills in a relaxed and informal online learning environment. Run primarily by graduate students, post-docs, and early career scientists, we recruit students who demonstrate a lack of access to conventional research experiences and connect them with the support they need to continue their education in plasma physics and fusion energy sciences. Over the 14-16 weeks of each year’s program, students are exposed to skills ranging from basic Python coding, to numerical methods, and finally advanced algorithms such as Monte Carlo methods and magnetohydrodynamics simulations using HPCs. This year concluded the program’s fourth cycle, and with it saw the largest class of 119 CRANE participants and at least 15 stipend-earning students. There has been a significant expansion of formal mentorship to include staff scientists and professors at premiere institutions. CRANE is working together with some of these institutions (UCLA, SAO, BMC, DIII-D, etc) towards creating formal internships, post-bac positions, etc for students who successfully complete the program. We present demographics, survey analysis, and key outcomes from the 2025 cycle of CRANE, updates from 2022-2024 CRANE alumni, and plans to expand for the 2026 cycle and beyond.

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Lessons Learned from Advanced Energy Technologies to Support a Robust Fusion Energy Supply Chain

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Thursday Parallel 1c - Setting Up Fusion: Supply Chain & Workforce Development, Kresge Little Theater (Building W16, downstairs), June 26, 2025, 10:30 AM - 12:00 PM

Fusion energy presents a significant opportunity to serve as a firm, scalable, emissions-free future energy supply resource. In addition to addressing technical and engineering hurdles on the pathway towards viable fusion pilot plants, establishing robust supply chains for (oftentimes novel) fusion materials and components is imperative for enabling commercial fusion power plants and sustaining a fusion power industry. Reducing supply chain risks is not a challenge unique to fusion; other emerging advanced energy systems (e.g., advanced nuclear reactors, concentrating solar power, thermal energy storage, and power plants with carbon capture and storage), along with established energy supply resources, content with supply chain challenges for deployment and sustainability. Lessons learned from adjacent energy technology supply chains can inform opportunities to support fusion energy. In particular, accelerating advanced materials and leveraging advanced manufacturing methods are common strategies across energy technologies, particularly advanced energy systems that have harsh operating conditions and aggressive deployment timelines.

EPRI is engaging energy technology developers, component manufacturers, and other supply chain stakeholders to identify needs, gaps, and solutions for advanced materials and manufacturing in advanced energy systems. EPRI has held three Supply Chain Workshops for Structural Components in Advanced Energy Systems, convening industry stakeholders across the advanced energy system supply chain landscape to identify challenges and opportunities, including processes for qualifying material and manufacturing methods, component testing infrastructure, and workforce development needs [1-3].

Both material and manufacturing process qualification may have long lead times or unestablished methodologies, and there are significant needs for capturing performance data in prototypic advanced energy system environment. Testing and qualification processes can be accelerated via collaborative industry programs for new materials development. Additionally, learning from test loops, pilot facilities, and other testing infrastructure will serve to provide practical experience with materials and identification of supply chain bottlenecks. Workforce development efforts that engage technical schools, universities, and industry groups locally, and globally, serves to bolster the development of supply-chain relevant skilled tradespeople, such as machinists and welders.
In addition to the Supply Chain Workshops, EPRI is developing, demonstrating, or qualifying multiple advanced manufacturing methods for energy technology applications, including:

- Powder Metallurgy-Hot Isostatic Pressing (PM-HIP)
- Directed Energy Deposition Additive Manufacturing (DED-AM)
- Powder Bed Fusion AM (PBF)
- Advanced Cladding Processes (e.g., diode laser cladding)
- Electron Beam Welding (EBW)

Insights from these efforts across energy generation options can be used to inform to fusion applications to support a robust supply chain of fusion-relevant materials and components.

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Perspectives on Public-Private Programs in the United States Context

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Thursday Parallel 1c - Setting Up Fusion: Supply Chain & Workforce Development, Kresge Little Theater (Building W16, downstairs), June 26, 2025, 10:30 AM - 12:00 PM

The challenge of developing a new energy source based on fusion power has led to dramatic growth in the private fusion industry in the last several years with nearly 40 companies in membership of the Fusion Industry Association. Many of these companies were founded out of public sector fusion science and technology programs (both national laboratory and university), and public sector investment and research and development continues to play an important role in moving towards the goal of realizing fusion energy. Starting with the Innovation Network for Fusion Energy (INFUSE) program in 2019, and moving into the milestone-based fusion development program announced in 2022, public-private partnerships have played a visible role in fusion development. This presentation will focus on the successes of current public-private partnership programs, and give a perspective on developing programs, such as the Public-Private Consortia Frameworks, the Fusion Innovation Research Engine (FIRE) Collaboratives, and the catalyzing influence of the reformulated Virtual Lab for Technology.

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Accelerating the Fusion Workforce in the USA

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Thursday Parallel 1c - Setting Up Fusion: Supply Chain & Workforce Development, Kresge Little Theater (Building W16, downstairs), June 26, 2025, 10:30 AM - 12:00 PM

The fusion energy research and development landscape has seen significant advances in recent years, with important scientific and technological breakthroughs and a rapid rise of investment in the private sector. The workforce needs of the nascent fusion industry are growing at a rate that academic workforce development programs are not currently able to match. This paper presents the findings of the Workforce Accelerator for Fusion Energy Development Conference held in Hampton, Virginia, United States of America (USA), on May 29-30 2024, which was funded by the National Science Foundation (NSF) of the USA. A major goal of the conference was to focus on bringing public and private stakeholders together to identify opportunities for partnership in fusion research and education with the goal of meeting the needs for a fusion workforce. Representatives from industry, academia, and national laboratories participated in the conference through the preparation of white papers, presentations, and group discussions, and the production of recommendations to address the challenges facing the fusion workforce in the USA.

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Spectroscopic analogue electronics for fusion neutron diagnostics

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Thursday Posters 1, Lobdell (Building W20 Room 208), June 26, 2025, 10:30 AM - 12:00 PM

The diagnostics for Deuterium-Deuterium (DD) and Deuterium-Tritium (DT) fusion neutrons is crucial for the operation of fusion reactors. Diamond detectors are one of the best candidates for fast and precise spectral neutron measurements. Their distinct response function to fusion neutrons, DD as well as DT, gives information on both, the temperature and spatial distribution of the fusion plasma. For spectral neutron analysis, charge amplifiers are the instrumentation of choice.
Due to high radiation fields it may be necessary to separate the detector and the amplifier using a detector cable. Charge amplifiers intrinsically increase their noise levels with the length of used detector cables, while decreasing the output signal. Any attenuation in the detector cable reduces the input signal in addition. These effects reduce the signal-to-noise ratio (SNR) and consequently the resolution of the measured spectra. A lower resolution spectrum limits the neutron measurement capabilities of the entire setup and therefore it is necessary to optimise the measurement setup.
In this presentation different approaches for the readout of diamond detectors with charge amplifiers will be discussed, their advantages as well as their limitations. The performance of various designs of charge amplifiers in fusion neutron diagnostics applications will be compared on the example of the DD-fusion neutron response function of a diamond detector.

Integrated Nuclear Design of a Fusion Pilot Plant

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Thursday Posters 1, Lobdell (Building W20 Room 208), June 26, 2025, 10:30 AM - 12:00 PM

Tokamak Energy is designing a Spherical Tokamak Fusion Pilot Plant (FPP) for integrated test and validations of technologies, systems and processes required for commercial fusion energy deployment. The FPP, which is targeting start of operations by 2035, will consist of an operationally relevant fusion environment. By exploiting the inherent plasma physics benefits of the Spherical Tokamak, the FPP will demonstrate scalable net power in a fully integrated system. Tokamak Energy and its FPP design efforts are supported by the U.S. Department of Energy's Milestone-Based Fusion Development Program.

Tokamak Energy's FPP, like any other DT fusion pilot plant concept, will be a radiation environment unprecedented in its harshness and complexity, owing to high 14 MeV neutron fluxes. Neutron and gamma radiation, and their effects, will permeate through and influence almost every component and system of a reactor. Moreover, these effects, and their mitigation, interact in complex ways in such a heavily interconnected system. Considering nuclear aspects too late in the design, or for systems and components in isolation, would result in a sub-optimally functioning plant. For a cohesively functioning and economically feasible power plant, neutronics and nuclear engineering, must thus be considered, carefully balanced, and integrated into the plant's engineering and design comprehensively from the very beginning and at every stage.

In this presentation, neutronics and nuclear engineering aspects of the global design of an FPP concept will be looked at through a top-down integrated lens. The influence of radiation effects on key systems and aspects of the plant, like material selection, power systems, fuel cycles, nuclear regulations, etc., will be explored. Additionally, knock-on interactions and trade-offs between these effects and their mitigation with other systems will also be considered.

The aim of this presentation is to understand, in an integrated manner, the complex neutronics interactions and trade-offs between various systems and other aspects, for the development of a commercially successful fusion power plant.

TOGA Device for Tokamak Fusion As It Should Be

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Thursday Posters 1, Lobdell (Building W20 Room 208), June 26, 2025, 10:30 AM - 12:00 PM

“Tokamak plasma should face only lithium” - this phrase is a guidance and an open “know-how” for the new approach to tokamak fusion. The toga invention (submitted as a PPA to the USPTO in Jan. 2025) is the upgrade to the Shafranov/Artsimovich tokamak [1], implementing this guidance. It introduces a hot (200-250C, Fig.1a) Wall with Continuously Creeping Liquid Lithium (24/7-CLiLiW, Fig.1b) for absorption [2] of escaping plasma particles, thus preventing plasma cooling by the uncontrolled hydrogen recycling. The combination of 24/7-CLiLiW with the high energy NBI (ENBI =60/120/180 keV for H/D/T isotopes) eliminates the cold particles in the system (S. Krasheninnikov, 1998, private communication).

In toga, the edge plasma temperature is comparable to the core temperature, Tedge $\simeq$ (1 − Rcooling)ENBI/5, the role of thermal conduction in the core is highly diminished, the Scrape off Layer is converted into a collisionless stream of energetic particles, directed into toga’s lower chamber with actively cooled heat sinks. Plasma Surface Interactions (PSI) are eliminated as a tokamak fusion topic. The escaping with the plasma ion temperature secondary charge exchange atoms H0 (representing 1/3 of the NBI fueling flux at standard ENBI) is not a concern for the LiLi layer, while the collisionless SoL is much simpler and more predictable than any plasma. With limited Li evaporation from the wall, and negligible H0 recycling or LiLi sputtering, the cooling coefficient Rcooling in toga is expected of the level of 0.2 or less (even with accounting for 3 plasma ion temperatures expelled by each evaporation Li atom absorbed by the plasma edge).

The minuscule flow rate 1 g/s (2 cm3/s) of 24/7-CLiLi (LZ, 2006) guaranties feasibility of its safely use in togas. For the future burning plasma 24/7-FLiLi is the only material capable to prevent accumulation of tritium in the vessel by continuously delivering T, dissolved in LiLi, to the outside processing and real time recovery. The burning plasma performance of the LiWF regime has been predicted astonishing [3], even with the cooling suppression by 0.5 at JET-like parameters. In fact, the correction of those calculations by using the standard ENBI for D and T as well as realization that the plasma electron component is not affected by recycling propels JET-size plasma beyond 1.5 x Lawson criterion, then converting ignition into a steady burn up.

The recent discovery of the plasma Li-doping effect (Fig.1c) on LTX-β (PPPL) opens the path for the mid-size toga devices (a $\simeq$ 0.5 m) for simulating the NBI-fueling with standard ENBI, relevant to the future burning plasma, despite some needs of controlled complementary (although cooling) ultrasonic gas or pellet injection. The existing TCV tokamak with a minor upgrade to a toga-like facility with the EAST-like Li coating by evaporation, can be a first candidate for making breakthrough toward tokamak fusion As-It-Should-Be.

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Design of Self-Cooled Lithium-Lead Fusion Blanket and Analysis of Tritium Breeding Performance with PHITS

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Thursday Posters 1, Lobdell (Building W20 Room 208), June 26, 2025, 10:30 AM - 12:00 PM

Maximizing the tritium breeding ratio, while taking into account the cost of lithium-6 enrichment is required to enable the commercialization of fusion energy. This study designed a spherical Self-Cooled Lithium-Lead (SCLL) blanket varying the thickness of the blanket and the enrichment of Lithium-6 in the lithium-lead breeder. The goal was to maximize the Tritium breeding ratio while accounting for the cost of lithium-6 enrichment. To reach this objective, an iterative methodology was adopted, alternating between design enhancements and rigorous simulations to refine the design approach. Per this analysis, 32.3 at% enrichment at a 134.9 cm blanket thickness is recommended for a cost optimized SCLL design to achieve a tritium breeding ratio of 1.3. This research on the intersection of design and economic analysis is a crucial step towards showing the economic feasibility of inertial fusion energy.

Discrete stellarator coil optimization with arbitrary spatial constraints

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Thursday Posters 1, Lobdell (Building W20 Room 208), June 26, 2025, 10:30 AM - 12:00 PM

Stellarator coils must produce strong, 3D-shaped magnetic fields with high accuracy. Due to these stringent requirements, the coils are one of the main cost drivers of stellarator devices. This contribution describes the development of a new coil optimization method that could help to find designs with convenient properties that could ease device assembly and reduce costs. In this method, coils are modeled as current distributions formed on a “wireframe”, or an interconnected mesh of current-carrying segments that encloses the plasma. This solution space enables the use of a discrete, greedy optimization procedure that entails adding loops of current to the wireframe one-by-one to construct and/or reshape coils [K. C. Hammond, https://arxiv.org/abs/2412.00267]. This method allows for arbitrary spatial constraints to be placed on where coils may form. Using these constraints, it is possible to develop coil designs that avoid collisions with other device components and keep the coils confined to user-prescribed regions that promote easy assembly and disassembly. Adopting such designs could reduce the time and costs required for reactor maintenance.

3D Neutronics Workflow and Performance Analysis for the HYLIFE-II Fusion Concept

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Thursday Posters 1, Lobdell (Building W20 Room 208), June 26, 2025, 10:30 AM - 12:00 PM

Inertial fusion energy systems are a promising path toward the achievement of clean and sustainable energy. However, developing a durable and efficient first wall remains a critical challenge. This work presents a workflow for designing and evaluating the impact of fast neutron fluence on the first walls of fusion engines, using the HYLIFE-II conceptual design as a test case. The HYLIFE-II conceptual design introduces a liquid wall solution to mitigate neutron fluence by utilizing a flowing liquid medium to shield the solid first wall. A simplified CAD model was created in SolidWorks, meshed in Coreform Cubit, and imported into the OpenMC Monte Carlo neutron transport code to assess key performance metrics, including displacement per atom, energy deposition profiles, and tritium breeding ratio (TBR). The TBR and energy deposition profiles were compared with a Serpent simulation to validate the viability of the workflow. In addition, a Python script was developed to extract simulation outputs and convert them into usable information for design performance analysis. The results provide further confirmation of the viability of the design in terms of breeding and material damage. The methodology and workflow developed in this work also show potential to be powerful design tools by allowing one to accurately predict power deposition, to pinpoint areas with above-average damage, and to single out zones where the addition of breeding material would be more effective.

Deterministic Neutronics for Stellarator Design

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Thursday Posters 1, Lobdell (Building W20 Room 208), June 26, 2025, 10:30 AM - 12:00 PM

Stellarators are complex 3D shaped fusion power plant candidates with inherent steady-state capabilities and no current-induced disruptions. Compared to the tokamak alternative, stellarators are thus more difficult to build and design but significantly easier to operate. Although its 3D shaping complicates design, it also allows for the optimization of the geometry for given parameters such as cost or size. Optimizing the geometry does
require a fast design cycle, including quickly assessing relevant engineering constraints.

However, engineering models for stellarators compatible with such a fast design cycle have only been introduced recently, and a blanket model that can quickly assess the neutronic viability of the design (e.g. tritium breeding ratio, fast neutron flux at the coils) is still lacking.

In principle, conventional Monte-Carlo methods could be used to evaluate the neutron response, but they can, especially in the 3D curved stellarator geometry, require significant computational resources and manual work, both in conflict with the desired fast design cycle. Furthermore, reduced models as developed for tokamak system-codes are incapable of modelling the highly heterogeneous neutron load in stellarators.

We have been developing a deterministic method for efficiently solving the neutron transport equation in (possibly parametric) stellarator geometry, providing a middle ground between expensive Monte-Carlo methods and inaccurate reduced models. It uses a discrete ordinates (SN), multigroup velocity space discretisation with an unstructured mesh,
arbitrary order, discontinuous Galerkin spatial discretisation.
This method has been tested in slab geometry using both analytical solutions and numerical benchmarks with the OpenMC Monte-Carlo code, including a three-dimensional breeding blanket test, showing good agreement in all cases while using significantly less computational resources. Results in parametric stellarator geometry will also be shown, including the use of novel stellarator-symmetric boundary conditions.

This mesh-based method is especially suited for design applications: the full distribution function is obtained on a computational mesh, facilitating informed design choices (e.g. adding shielding in which region), and straightforward coupling to thermomechanical codes. Furthermore, the deterministic nature of the model together with the low computational cost can enable design optimization applications.

Century: Zap Energy's 100 kW-Scale Liquid Metal Lined and Cooled Repetitive Sheared-Flow-Stabilized Z-Pinch Fusion Technology Demonstration Platform

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Thursday Posters 1, Lobdell (Building W20 Room 208), June 26, 2025, 10:30 AM - 12:00 PM

The sheared-flow-stabilized (SFS) Z-pinch concept is on a path to commercialization at Zap Energy. [1] The SFS Z pinch relies on plasma self-organization which eliminates the need for external confinement or heating technologies. This pulsed, compact magnetic confinement technology could, in turn, provide the basis for a cost-effective deuterium-tritium fusion power plant. [2] In addition to a robust experimental program pushing plasma performance towards breakeven conditions, Zap Energy has parallel programs developing power handling systems suitable for future power plants. Technologies under development include high-average-power repetitive pulsed power, high-duty-cycle cathodes, and liquid metal wall systems. Century is Zap Energy's first platform integrating these three components into an operational system capable of firing SFS Z-pinch plasmas into a liquid-metal-lined container at sustained repetition rates on the order of 0.1 Hz. The pulsed power driver and liquid metal heat exchanger are both designed for a nominal average power of 100 kW. Results from the first year of operations are described.

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Effects of magnetic fields on the formation and migration of CLF-1 steel corrosion products

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The CLF-1 steel, the main candidate structural material for the cooling water pipes in the fusion reactor blanket being developed in China, will be corroded by the cooling water in the fusion blanket at high magnetic fields. The magnetic field can be as high as 14 T. The effect of the magnetic field on the corrosion process of CLF-1 steel should be considered. To study the influence of the magnetic field at the scale of 10 T on the corrosion process of CLF-1 steel, corrosion experiments were carried out using a superconducting steady-state magnet with a 14 T high magnetic field at room temperature for 2 h. The Experimental Advanced Superconducting Tokamak (EAST) can generate a magnetic field environment similar to that of the fusion reactor. Water corrosion experiments were also carried out in the magnetic field environment of EAST using the same corrosion solution to the experiment in 14 T magnetic field for 2 h, 12 h and 24 h. The magnetic field strength of EAST at sample position was measured using a Tesla meter at the time of the corrosion experiment, which was between 21 mT and 85 mT. The solution was deionized water with 1wt% H2O2 + 0.1wt% NaCl to accelerate the corrosion process. Contrast experiments without the magnetic field were performed. The ANSYS software was used to simulate the distribution of magnetic flux around the sample in a 14 T magnetic field. The morphology of the samples was observed and the composition of the corrosion products was analyzed. The number of corrosion pits on the sample surface increases significantly at 14 T magnetic field and shows almost no change at EAST field. The surface corrosion products around the pits appear approximately circular in shape at 14 T and EAST magnetic fields, while the corrosion products distributed in strips on the sample corrode in the absence of magnetic fields. The simulations show that the maximum magnetic flux density is localized in the middle of the sample at 14 T, where most of the corrosion pits are distributed. Possible reasons for the influence of the magnetic field on the corrosion products distribution could be that the Lorentz force and magnetic gradient force induced change the motion of ions and paramagnetic substances (such as O2) respectively.

Keywords: Magnetic field, Reduced activation steel, Corrosion, Corrosion Products

High Field Compact Machine Approach to Fusion:Main Technology Issues

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High Field Compact Machine Approach to Fusion: Main Technology Issues*

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Among various options being considered for the objective of proving the scientific feasibility of a meaningful fusion reactor, the line of high magnetic field compact machines [1] remains the most promising, given its cost effectiveness and its demonstrated ability to produce well confined and pure plasmas with record high pressures.

The relevant design criteria involve maximizing the poloidal magnetic field (Bp) together with the plasma current (Ip) that can be sustained under macroscopically stable conditions with a feasible aspect ratio for the plasma column. The plasma (DT) reactivity and the maximum plasma pressure have a strong dependence on Bp while a high plasma current ($Ip \propto a\ Bp$) is important for the needed energy confinement time.

The most critical issue in the machine design is that the plasma chamber has to survive the disruptions of large plasma currents (e.g. 10 MA) immersed in high toroidal magnetic fields. In the case of the Ignitor machine [2], this has led to the adoption of an inhomogeneous (poloidal) thickness of the plasma chamber that reaches a considerably high value ($\simeq$ 6.5 cm). The mechanical structure that can be adopted to optimize the stress distribution within the toroidal magnet is of the “bucking and wedging” kind and involves a mechanical coupling of the toroidal magnet with the central solenoid and of this with the central post. A containment ring and a set of C-clamps to withstand the vertical stress components complements this solution. The present, the fully designed version of the Ignitor machine involves high temperature superconducting (HTS) magnets for the largest poloidal field coils but maintains the “copper solution” for the highest field magnets given the difficulty of adopting a highly reliable quench protection system for the available superconducting materials. Considering the attractive properties of HTS materials, a significant effort is being devoted to dealing with this issue in the context of the Ignitor program and existing international collaborations.

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Achievements and challenges of the first experimental campaign at the SMall

Aspect Ratio Tokamak (SMART)

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The SMall Aspect Ratio Tokamak (SMART) [1] is a spherical tokamak (ST) [2] operated by the University of Seville. Its primary objective is to investigate the potential combination of Negative Triangularity (NT) [3] and low aspect ratio as a viable scenario for a future compact Fusion Power Plant (FPP). SMART has been designed to operate at $Ip \leq 1MA$ and $Bt \leq 1T$ with the magnetic axis placed at Rmag= 0.45m in the baseline scenario. The poloidal coil system of SMART allows flexible shaping capabilities with aspect ratios $1.46 < A < 3.0$, elongations $k < 3.0$ and triangularities $-0.6 < d < 0.6$. In the first operational phase, ohmic plasmas are explored while Neutral Beam Injection (NBI) heating is planned for phases 2 and 3. Scenarios for the different phases have been designed using a combination of FIESTA and TRANSP codes. While FIESTA provided the equilibrium and the coil current evolution for a specific plasma scenario, TRANSP used different transport models to predict the machine performance in the different phases. This contribution provides an overview of the machine and the achievements of the first experimental campaign. Furthermore, challenges encountered during the experimental campaign are discussed.
The advantages of the ST-NT concept initially explored at SMART are extrapolated to a FPP based on this scenario. Initial considerations for the operational point, taking into account engineering constraints, are presented. Examples of plasma equilibria obtained with a preliminary coil system are given. The advantages of ST-NT in providing a solution for the blanket design, neutronics and power exhaust are discussed. Predictions of plasma performance obtained using the ASTRA transport code are also described.

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Application of ASME sec.VIII div.2 to manufacture nuclear pressure equipment of ITER under the ITER module H1

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ITER is a nuclear basic facility INB-174 in France which requires to apply French regulations for manufacture of nuclear pressure equipment. This French regulation is provided in the French environmental codes so called L557 and R557 as well as in the ESPN order and Pressure Equipment Directive (PED). In the sense of this regulation, code and standard is a tool to meet the Essential Safety Requirements (ESR) of ESPN order and PED. As of today, there are several harmonized codes to PED but no harmonized code to ESPN order. It is sole responsibility of manufacturer to demonstrate how to satisfy the ESR for the conformity assessment.
Applying international standards for manufacturing French nuclear pressure equipment has become to be keen in a sense of globalized supply chain of nuclear industry. As the ASME is one of well-known manufacturing codes of pressure equipment, ITER used this ASME section VIII division 2 to manufacture six different types of nuclear pressure equipment whose ESPN level is N3. As ITER has developed the module H1 which is a module for conformity assessment by using its own quality assurance program, those vessels follow the ITER module H1.
This paper is to provide the practices how ITER applied the ASME to conform the ESR of French regulation by explaining the major points what ASME does not cover from design, procurement, fabrication and transversal points in different types of vessels. In addition, this paper suggests the considerable points when manufacturers use their own module H1 for the conformity assessments of French nuclear pressure equipment.

Computational Study of a Cryopump Design for Fusion Exhaust Gas Purification

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The primary goal of the inner loop of the fuel cycle in a fusion reactor is to enable a low fusion fuel inventory while continuously removing impurities from the reactor exhaust gas prior to directly circulating fuel to the fuelling injectors. The concept of recirculating fuel through an inner loop of the fuel cycle is called the direct internal recycling (DIR). A typical fusion exhaust stream contains many trace hydrocarbons and other gases introduced to sustain edge plasma conditions along with the helium ash and unburned fuel. Thus, a successful DIR concept should include components to remove impurities from the exhaust so that the purified fuel can be reintroduced into the reaction chamber. A cryopump is a good candidate to remove impurities from the exhaust through cryo-absorption. Ideally, a cryogenic pump system should have a continuous gas pumping with a desirable pumping speed that maintains a continuous recirculation of fusion fuel and a low divertor neutral pressure.

In this study, we investigate a conceptual cryopump, that would be the first stage of a continuous cryogenic DIR concept, through computational methods. The conceptual design consists of several sets of chevron-fins maintained at various cryogenic temperatures targeting the saturation temperature of different exhaust gases. The pump is designed to cryotrap impurities before the exhaust feeds into another continuous cryopump segment which separates helium from the fusion fuel.

The computational study is performed using the direct simulation Monte Carlo (DSMC) method to facilitate design of a lab scaled experimental cryopump. The DSMC results help characterize the effectiveness of the cryopump in removing typical fusion exhaust gas impurities. Additionally, the overall pressure-drop and heat transfer in the pump from the simulations provide information on the cooling and pumping requirements for the experimental setup. This paper will highlight some key observations of the computational study.

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Determination of the Effects of Plasma Shape on the Electromagnetic Loads during Plasma Disruptions

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During plasma disruptions in a tokamak, the plasma becomes unstable and losses thermal energy confinement and the current decays to zero, potentially damaging the reactor from thermal and electromagnetic loads. With the sudden change in the plasma current, there is a large, rapid change in the poloidal magnetic field, causing eddy currents in the components of the reactor. These eddy currents interacting with the magnetic fields generated by the field coils can impart Lorentz forces upon these components. As a negative triangularity plasma shape has been shown to be promising in mitigating edge-localized modes (ELMs), determining the effect of plasma shape on electromagnetic loads during plasma disruptions is crucial for modeling realistic scenarios. Additionally, a current quench can either be modeled in linear or exponential form; the variation between each form of current quench can result in a different rate of growth in the induced currents and electromagnetic loads. New plasma equilibria are constructed using FreeGS free boundary equilibrium code for a set of user-defined plasma geometries and profiles with desired triangularity within a tokamak fusion pilot plant. The plasma geometries are fed to the reactor model within the Ansys Maxwell code to calculate the induced current and Lorentz forces on the blanket during a midplane disruption. This approach can help to quickly understand the effect of plasma disruptions on in-vessel components and improve design iterations.

Summary of recent physics results from LTX-beta

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The Lithium Tokamak Experiment beta (LTX-beta) is the only tokamak with metallic plasma facing components (PFCs) that are fully coated in lithium[1]. The device was maintained under ultra high vacuum conditions for 5 years for its previous campaign. During the campaign two iterations of lithium evaporators were tested and evaluated. A total of 60 grams of lithium was deposited on the PFCs. The tokamak was operated safely and continuously with both solid and liquid lithium PFCs. Key physics results from the campaign will be presented, including the observation of neutral beam heating with solid lithium PFCs [2]. Discharges were produced with flat electron temperature profiles lasting multiple energy confinement times[3]. Lithium coatings reduced the recycling coefficient to a record value of 0.5 with both solid [4] and liquid lithium. And, finally, both beam heating and fueling was demonstrated with liquid lithium walls. Low temperature gradients and low recycling were concurrently observed with a low collisionality, $0.01< \nu^{*}< 0.1$, scrape-off layer (SOL). These results have important implications for other fusion experiments such as ST-40 and NSTX-U that intend to operate with fully coated lithium walls. These implications will be discussed. The device was vented following the five year campaign to adjust beam position to allow beam operation at higher power, perform general maintenance, and upgrade the diagnostics as well as the lithium evaporators. LTX-beta is now operating with a primary focus on liquid lithium operations. The upgraded diagnostic suite will include new re-entrant photodiode based Lyman alpha and soft X-ray arrays to diagnose recycling and tearing mode activity at lower collisionality. An inboard limiter and infrared imaging system will diagnose the power deposition profile as a function of edge collisionality. A new edge Thomson scattering system will characterize the low collisionality SOL. Evaporator upgrades feature between-shot remotely actuated and loaded evaporators that will form the design basis for ST-40 and NSTX-U lithium evaporators. Details of the upgrades will be presented along with any new results from the current and final campaign.

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Thermal analysis and optimization for ITER EP09 DSM2

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Diagnostic Shielding Modules (DSM) are immediately behind the plasma facing Diagnostic First Walls (DFW) and provide support to them. They are the secondary containers of the ITER diagnostic Port Plugs where shielding and diagnostics components must be integrated together and to protect the diagnostics components from plasma radiation while providing apertures for diagnostic access to the plasma. For the development of Equatorial Port DSMs, several key requirements have to be met including structural, thermal, electromagnetic and inertial loading and the combinations. This paper is about the thermal analysis modeling, result and recommended design modifications to meet the requirements.

Each equatorial port has three DSMs and the DSM2 was subject to the highest heat load including nuclear heating from gamma ray and neutronics and surface heating from plasma thermal radiation. There are three main thermal scenarios for design verification: normal operation full load, normal operation pulse load of 25% duty ratio and baking. The ITER Research Plan foresees different design schemes for the cumulated D–T plasma duration. This paper uses two of them, full load steady state and pulse load transient with 25% duty cycle, to simulate thermal behaviors during normal operation. Baking is to heat all the component to high temperature, 180 ^{o}C in this model, to remove impurities and achieve a high-quality vacuum. A solid-fluid integrated model was built in ANSYS MAPDL for these thermal simulations. Flow distribution of the water supply was calculated by ANSYS CFX. A few sub models were also used to determine the best water supply ratio and optimize water channel design, which provide a guidance to orifice dimension selection in the fluid model. Then the water flow rates from the fluid model were applied to the fluid elements in the solid-fluid integrated model. Surface elements with convection to fluid elements were created on the channel surfaces and their heat transfer coefficients depend on fluid temperature, conductivity, and viscosity etc. Thermal contact resistances are carefully modelled and consistent with other ITER thermal models. The solid-fluid model for normal operation and baking are similar, except for the flow rate, inlet temperature of fluid elements, and heat transfer coefficient for the surface elements on channel surfaces. Heat loads during normal operation include nuclear heating and surface heating from plasma thermal radiation. The same model can be used to run both static and transient cases. The results verified the working temperature ranges of each component to be within allowable. Baking requirement of reaching 180 ^{o}C in 48 hours could be met. The cooling process was simulated to show the final temperature distribution after 24 hours. Temperature maps from this analysis were generated and provided for following structural analysis. Temperature history maps were also provided to other tenant component modeling as boundary conditions.

VmecHub – A platform for 3D MHD equilibrium data at W7-X

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Magnetohydrodynamic (MHD) equilibrium calculations are of central importance for scientific research on fusion using magnetic confinement. For example, for the design of a new stellarator machine, MHD models can be used to evaluate the complex three-dimensional (3D) coil geometries and thus optimize them based on certain physical properties. During experimental operation, equilibrium calculations are the basis for the analysis of diagnostic data. The Variational Moments Equilibrium Code (VMEC) is a code often used for this purpose in the stellarator community. Since it provides a numerical solution for 3D geometries, the code is significantly more complex and the computing time is longer compared to 2D calculations for tokamaks.

As consistent equilibrium calculations are crucial for experiments, a web service for VMEC calculations has been implemented for the Wendelstein 7-X (W7-X) stellarator. This web service can be used to start new VMEC calculations on a server and to retrieve the results of completed calculations. It has been successfully in operation for several years. During this time, its features were continuously expanded and improved. In the process, this service has evolved into a central platform for MHD equilibrium data: VmecHub.

The VmecHub platform provides a repository and data catalog for VMEC data at W7-X. The goal of the platform is to support the entire data life cycle, with each step being manageable by users via self-service. The data is easily accessible and discoverable for the entire W7-X team, via website or programmatically via APIs. Each calculation is accessible via a unique identifier and associated URL, which supports collaboration and data sharing, thereby contributing to traceability of results in the sense of good scientific practice.

Data discoverability becomes more and more important with a growing data collection. Therefore, a variety of features has been developed to support users in their exploration. These include advanced search functions that allow users to search for suitable equilibria based on physical properties or on specific experimental data. The platform provides a web page for each VMEC calculation that displays its key parameters in a concise form. This includes physical properties and metadata (e.g. origin, links, etc.), as well as plots of the resulting flux surfaces and other values. The platform also allows users to curate a collection of equilibria on a specific topic.

This contribution presents the key features and software architecture of the VmecHub data platform.

Utilization of neutral beam energy fractions in motional Stark effect spectra for radial electric field measurements in KSTAR

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Thursday Posters 1, Lobdell (Building W20 Room 208), June 26, 2025, 10:30 AM - 12:00 PM

A motional Stark effect (MSE) diagnostic is crucial for the measurement of the internal magnetic field in tokamaks by analyzing the light emitted from neutral particles injected into the plasma. MSE diagnostics utilize the Stark polarization and splitting of emission from the interaction of neutral particles and plasma, under the Lorentz electric field induced by the neutral beam velocity and the local magnetic field. Previous analyses of MSE data assumed that the Lorentz field is the only electric field contributing to the Stark polarization and thus that the direction of the magnetic field could be directly determined. However, it is difficult to accurately measure the magnetic field component since the radial electric field inherently formed inside the plasma is mixed with the Lorentz field. This work describes the approach to derive the radial electric field by measuring and comparing the polarization angles of the full- and half-energy components of the neutral beam injected using the background polychromator-based MSE (MSE-BP) in KSTAR. Since the measurement wavelengths of the two components depend on the beam energy, we calculated the wavelengths in advance, selected appropriate filters and then installed them in the MSE-BP system. The polarization angle measurements were performed for various toroidal field values with intra-shot pressure variations over time. Additionally, measurements were performed by varying the filter position(offset) under the same conditions. The polarization angles obtained from all cases using MSE-BP were consistent with those measured using the KSTAR-MSE system, and variations due to pressure changes as well as differences related to the magnetic field were observed.

Computational Modeling of the Lobo Lead Loop with MOOSE: Predicting Lead-Lithium Behavior in a Magnetic Field

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The Lobo Lead Loop (LLL) at the University of New Mexico (UNM) is one of the key test beds for corrosive experiments using molten lead, a very important material being considered for advanced fusion reactors. This study applies the Thermal Hydraulics Module presently under development within the MOOSE framework to simulate the complex fluid dynamics and heat transfer phenomena within the LLL. By replicating the geometry and operating conditions of the loop, we benchmark MOOSE predictions against experimental data obtained from the LLL. This validation is critical for establishing confidence in MOOSE's ability to ultimately model lead-lithium behavior, a promising tritium breeding material for Dual Coolant Lead Lithium (DCLL) fusion reactor blankets. Accurate simulation requires careful consideration of temperature-dependent thermophysical properties of lead, including density, specific heat, thermal conductivity, and viscosity. It includes applying detailed boundary conditions for flow rates at the inlet/outlet, pump pressure, and heat fluxes to model the experiment with fidelity. The simulation considers all the heat transfer mechanisms involved and correctly predicts temperature distribution and pressure drop inside the loop. The present LLL does not have a magnetic field, and therefore, it has limited applications for magnetohydrodynamic studies. However, a new loop with lead-lithium as the working fluid and with a magnetic field of 1 to 2T is currently under development, and this MOOSE framework will be extended in the future to simulate the behavior of the new loop by including modules such as THM, Heat Transfer, and Electromagnetics to predict the impact of the magnetic field on fluid flow and heat transfer within the loop, and possibly also assist in the design of the new loop.

Spatial Profile Measurements of Te and ne using a Reciprocating Langmuir Probe and Optical Emission Spectroscopy

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The Hybrid Illinois Device for Research and Applications (HIDRA) has previously been reported to operate with electron temperatures as high as 20 eV and electron densities of 1e18 m^(-3). . Under stellarator mode, HIDRA's plasma has what is known as a hollow temperature profile and shows to consist of detectible fast electrons. For future experimental usage of conducting plasma-material interaction research, it is important to understand plasma properties and behavior. A Reciprocating Langmuir Probe (RLP) is used to verify the spatial plasma parameters in HIDRA. The process is performed using He and H2 plasmas at 20%, 50%, and 90% of 6 kW magnetron heating (at 2.45 GHz), under pressures of 4e-5, 3.5e-5, and 2.9e-5 Torr. Optical Emission Spectroscopy (OES) is a secondary diagnostic used to verify the plasma parameter data by applying the Collisional Radiative Model (CRM). At different operating environments and conditions, the electron temperatures obtained from the probe hover around 14 - 17 eV at the core and variations between 22 - 30 eV at a position 4 cm from the core, illustrating a hollow temperature profile as was initially assumed. The electron density is found to be around 1e18 m^(-3) and shows consistency with the expectation. There is no distinction between different operating pressures and power due to the interference of the background gas in HIDRA. Future presentation will include the results obtained from the spectroscopy CRM to further verify the accuracy of the data derived from the Langmuir probe as well as the calculation for the fast electron population.

Developing a Pressure Relief Philosophy for Tritiated Systems: Hydrogenation Kinetics of Depleted Uranium and its Application for Emergency Recovery of Tritiated Fluids

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The global distribution of fusion power plants has the potential to provide abundant, low-carbon energy, playing a transformative role in addressing climate change. As fusion technology progresses from experimental rigs to industrial-scale facilities, tritium handling systems must be adapted to manage larger quantities of hydrogen isotopes while ensuring fuel cycle availability. Current designs rely on over-sized equipment, multi-layer containment held at sub-atmospheric pressures, and emergency shutdown systems to mitigate risks of high-pressure scenarios in tritiated systems [1]. However, for commercial fusion power plants, such approaches are financially unsustainable.

This study aims to develop advanced control strategies and a robust pressure relief framework for tritiated systems to enhance fuel cycle safety, reliability and economic feasibility. Depleted Uranium (DU) metal getter beds, a leading technology candidate for long-term hydrogen isotope storage, offer significant advantages in pressure relief designs. DU beds enable rapid separation of hydrogen isotopes from non-hydrogenic species allowing for swift recycling of hydrogen isotopes within the fuel cycle with limited process disruption. Additionally, DU beds store isotopes at lower containment pressures, reducing the downstream pressure of a relief device, which enhances safety and lowers the equipment design requirements. This, in turn, minimises construction costs, tritiated waste, and system tritium inventory.

This work investigates over-pressure scenarios, such as loss of cryogen accidents (LOCA) in the cryogenic distillation columns used for hydrogen isotope separation. Steady-state models quantify pressure relief system requirements for worst-case scenarios using bursting disks and expansion vessels. Building on this, dynamic models are developed to evaluate the absorption rates of hydrogen isotope mixtures onto an emergency DU storage bed. These models inform optimal DU bed integration within pressure relief systems, capturing hydrogenation kinetics and predicting pressure transients within the cryogenic distillation columns during LOCAs. The results provide a foundation for designing effective pressure relief strategies, enhancing the safety and viability of commercial fusion fuel cycle systems.

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Approximate Geometric Representations and Mesh Adaptivity for Deterministic Transport Solvers in Fusion Neutronics

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High-fidelity 3D neutronics analyses play a critical role in assessing a broad range of design parameters for any given fusion system. Traditionally, Monte Carlo methods have been relied upon as the standard tools for most problems in this space due to their inherent flexibility and ease of use but are often incapable of producing fast and fully converged solutions to a variety of thick-shielding and deep penetration problems that come up in the engineering design of fusion devices. Deterministic neutron transport methods can directly provide fine-grained spatial and angular flux distributions globally but are much more difficult to link directly into the engineering design process due to the difficulty of preparing fully compatible CAD assemblies due to mesh contiguity requirements.

A mesh construction and processing algorithm built around Silver Fir Software's Cottonwood and Poplar utilities is presented here. This algorithm processes an octree mesh constructed by the Poplar utility from an input MCNP model and efficiently subdivides a discontinuous forest of octrees into a contiguous tetrahedral mesh with exact region boundary representations. The resulting mesh retains the underlying goals of the Poplar meshing routine; adequate resolution to capture the transport solution, high accuracy of overall material/part interfaces, and reduced refinement wherever high refinement is not needed to produce an accurate transport solution. This mesh reconstruction routine allows for the construction of efficiently adapted tetrahedral meshes of MCNP models into Attila-compatible forward transport models. Mesh adaptation can be fine-tuned with the pre-existing heuristics within Poplar and supplemented with minor geometry modifications and additional region-wise or material-wise heuristics during the initial construction of the MCNP unstructured mesh geometry to optimize model performance. The workflow is fast, efficient, and leverages the full capabilities of the Attila solver without significant additional design support overhead in preparing CAD for analysis.

This method was used to construct approximate models of the SPARC tokamak facility under construction by Commonwealth Fusion Systems from unstructured mesh models used in MCNP workflows. The approximate models show remarkable ability to replicate the spatial and energy flux distributions of the exact-representation Monte Carlo model in the near-field, while also being fully capable of producing rapid solutions to deep-penetration and far-field problems that have proven intractable with other methods. This work demonstrates that multi-level octree-based approximate geometries can reach the necessary solution fidelities to serve as a primary neutronics analysis workflow for problems that are not feasible in Monte Carlo tools.

Conceptual Design of Solid Breeder Blankets for Fusion Pilot Plants

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A four-stage approach to Indian DEMO has shown that a gross electricity producing fusion pilot plant is absolutely essential to bridge the technology gap between the present-day devices including ITER and a net electricity producing DEMO reactor [1,2]. One of the crucial technological gaps while extrapolating from ITER to DEMO is the demonstration of high-grade heat extraction and power conversion, and tritium breeding, extraction and its re-use from the breeding blankets. The pilot plant design exactly aims to address these issues on a smaller scale to ensure the technology readiness for the credible extrapolation to DEMO and power plants.

India has already been developing breeding blanket concepts under the ITER test blanket module (TBM) frame work using both solid and liquid breeder materials. The lead lithium ceramic breeder (LLCB) concept is envisaged to use molten Pb-Li as a coolant and neutron multiplier flowing around the canisters of ceramic breeder Lithium meta- titanate (Li2TiO3) pebble bed [3]. In helium cooled solid breeder (HCSB) blanket, pebbles of Li2TiO3 will be used as the breeder material and beryllium is the neutron multiplier [4]. Two different types of the stacking, radial and poloidal, of alternating layers of breeder and multiplier materials separated by cooling plates were studied for the HCSB concept [5,6]. Yet another concept using titanium beryllide (Be12Ti) as neutron multiplier with cooling tubes with a mixed pebble bed of breeder and multiplier has also been investigated [7].

In this paper first we present the comparison of different concepts in the context of a fusion pilot plant of 3.6 m major radius, 300 MW fusion power with a neutron wall load of 0.75 MW/m2. We further discuss the case where the entire outboard is covered with helium cooled breeding blanket modules mounted on to a frame. The heat extraction from the breeding blanket modules and the tritium production in the modules will be discussed. The thermal response of the blanket for a fusion pulse length of 3000 s will be presented. The blanket performance will be evaluated using CFD, thermal and structural analysis using ANSYS. DEM-CFD simulations to validate and estimate the effective thermal conductivity of mixed bed will also be discussed in this paper. Impact of inlet temperature on the operation scenarios for optimizing the coolant flow rate will also be presented. Purge gas flow analysis for tritium extraction efficiency will also be presented along with the estimate of average tritium breeding ratio.

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First wall nuclear heat modeling from a beam injected deuterium-deuterium fusion neutron source in OpenMC

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Eos will be a sub-breakeven, beam-target, deuterium-deuterium fusion, planar-coil stellarator. Beam-target fusion introduces distinct energy, spatial, and anisotropic angular distributions in the neutron source compared to thermonuclear fusion systems. In this work, we present efforts to model the Eos beam-target neutron source in OpenMC and analyze the resultant nuclear heating on a simplified model of the first wall using the cell-under-voxel method. To model the beam-target source, characteristic information of a discrete number of source sites is precomputed using plasma equilibrium information provided from DESC. A custom neutron source function for OpenMC loads the source site data at runtime and uses random samples to select the neutron source site and calculate the initial energy and direction of travel. Using the cell-under-voxel method in OpenMC, significant neutron peaking is observed in the first wall due to the toroidal and poloidal variation of the stellarator and the anisotropic emission of neutrons. Insights obtained from these results will have significant implications for future Eos first wall, blanket, and shielding designs.

Exploring CSG-to-CAD Workflows using Experimental Benchmarks for OpenMC code validation

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The immense complexity of fusion reactors is driving nuclear analysts to seek tools capable of incorporating Computer Aided Design (CAD) models into their workflows. However, integrating CAD geometries into neutronics codes is not only challenging and fragmented but also demands rigorous validation. Still, almost all the experimental benchmark models available for validation rely on classic constructive solid geometry (CSG).

The study evaluates multiple workflows for converting CSG models to CAD representations. This work investigates various CSG-to-CAD conversion workflows to identify the most effective approaches for neutron transport simulations using OpenMC. The validation is performed using the fusion neutronic shielding benchmark from the Oktavian experiment.

CAD models are converted into a format suitable for Direct Accelerated Geometry Monte Carlo (DAGMC) by tessellating the surface geometry into high-quality unstructured meshes (UM) composed of triangular elements to enable detailed simulations.

Results are obtained using both fully open-source software stacks and workflows incorporating proprietary tools, highlighting their respective strengths and limitations.

A comprehensive comparison is conducted between the results obtained from CSG-based and CAD-based simulations in OpenMC, as well as MCNP® CSG-based results and experimental benchmark data provided by the Compilation of Nuclear data Experiments for Radiation Characterisation (CoNDERC). The accuracy of the measured leakage neutron and gamma spectra from sphere composed of different materials is assessed across all workflows, providing insights into the reliability of CSG-to-CAD conversions for nuclear applications.

Additionally, a performance assessment of the computational codes is presented, emphasizing the efficiency and scalability of different approaches. Uncertainty quantification (UQ) is performed to evaluate the impact of input cross-sections data uncertainties on the simulation results, understanding how these propagate to the final outcomes. The comparison reveals strong consistency with experimental data, and in some cases, the results obtained show better accuracy than those reported in the literature using other Monte Carlo codes. This confirms the reliability of OpenMC for neutron transport simulations in fusion environments.

Demonstration of a Continuous Cryopump for Purification of Fusion Fuel for Direct Internal Recycling

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Thursday Posters 1, Lobdell (Building W20 Room 208), June 26, 2025, 10:30 AM - 12:00 PM

Future DT fueled fusion power plants must breed fuel and sustain a burning plasma using a semi-closed loop fuel cycle. The DT fusion fuel cycle's purpose is to provide fuel to the plasma, pump and separate machine exhaust products, and recover fuel from breeding and plasma exhaust products. Current fuel cycle technology requires a large building for palladium membranes and cryogenic distillation columns which results in a large inventory due to long processing times. The concept of directly recirculating the exhaust gas, bypassing the tritium plant, to make fuel pellets was proposed in the 1990s and later termed Direct Internal Recycling (DIR). In the DIR concept, the residual fusion fuel in the machine exhaust stream is separated locally and diverted directly to the fueling systems, bypassing isotopic separation and other processing equipment, and therefore significantly reducing the required size of the fuel processing plant, reducing plant inventory, and thus increasing the economic viability of fusion as an energy source.
One concept for DIR consists of a series of cryogenic pumps to separate the impurities from the machine exhaust gas by utilizing the different triple point temperatures of exhaust constituents. In this concept, the plasma exhaust is initially passed through an impurity trap operating at ~25-30 K to desublimate impurities such as hydrocarbons, argon, oxygen, and nitrogen. The resulting process stream will consist of DT fuel and helium. The process stream is then pumped by a continuous cryopump known as a "snail pump". This pump is a steady state continuous cryopump that desubliminates all remaining exhaust gas constituents while allowing helium, a byproduct of the fusion reaction, to pass through. The helium is pumped to the tritium plant for processing while the desublimated material is continuously scraped off, heated up, and transported to the fueling system. This paper will outline the design and operation of the snail pump, along with results on separation efficiency versus inlet flow rates.

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Upgrades to the Thomson scattering diagnostic on the Lithium Tokamak eXperiment-beta (LTX-β)

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We present the upgrades to the Thomson scattering diagnostic on the Lithium Tokamak eXperiment-beta (LTX-β) to provide complete radial coverage of the plasma. LTX-β is a spherical tokamak with a lithium-coated low-recycling plasma-facing shell conformal to the last closed magnetic flux surface [1, 2]. Flat temperature profiles with peaked density were observed in LTX-β [3] using an existing multipoint Thomson scattering (TS) diagnostic [4] with limited radial coverage at the plasma core. There is a need for more comprehensive core and edge Thomson scattering (TS) measurements to thoroughly examine beam fueling, beam heating, and transport under low recycling conditions with liquid lithium walls and in a lower collisionality scrape-off layer (SOL) [5, 6]. The LTX-beta device is uniquely equipped to investigate these phenomena. Therefore, we have redesigned the old high-field side (HFS) region TS setup and added a new TS setup in the scrap-off-layer (SOL) region with five new "lines-of-sight" to explore these profiles further with broader radial coverage from R=15 cm to 56 cm. The new HFS TS system will provide profile measurements from a major radius R=15 to 45 cm at 10 radial points, providing a full electron temperature profile, as compared to the old HFS system covering R=20 to 40 cm at five measurement points. Additionally, the HFS TS setup is equipped with faster light collection optics, which will be used with a redesigned fiber bundle set to image the scattered light at a scattering angle of 150 degrees with respect to the laser line as compared to the old setup working at 80 degrees; we anticipate this change will lead to a significant increase in signal strength. As a result, we expect to confirm that the electron temperature profile on the inboard side of the discharge axis is flat for low recycling lithium walls and reduce the number of shots needed to document a profile.

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Study of the high-energy background in gamma diagnostic for fusion power measurements in DT tokamaks

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Measurement of fusion power in a tokamak is essential for operations control, licensing, and research. Traditionally in DT fusion plasmas, fusion power is measured by counting neutrons produced by the $T(D,n)^4He$ reaction. However, a novel method based on gamma-ray spectroscopy of the $T(D,\gamma)^5He$ reaction has been successfully demonstrated at JET[1,2]. While neutron diagnostic requires the calculation of the adjoint flux using an accurate MCNP model of the reactor, gamma-ray diagnostic only requires knowledge of the plasma volume in the detector line of sight and of the plasma spatial distribution: furthermore, it does not require extensive in-vessel calibration campaigns.

Although promising, this technique presents several challenges, such as developing an appropriate neutron attenuator, which is mandatory since the $T(D,\gamma)^5He$ reaction branch is $2.4\cdot10^{-5}$ less likely than the $T(D,n)^4He$ branch and obtaining precise simulations of background, especially in the high-energy region around the 16.7 MeV peak of the $T(D,\gamma)^5He$ reaction.

To properly understand what is causing the background in the energy range between 15 and 20 MeV, beam-on-target experiments were performed at the Frascati Neutron Generator (FNG). MCNP simulations allowed to understand the origin of the background at high energies that resulted to be caused almost entirely by steel and copper in the proximity of the target. The finding is consistent with other studies found in literature [3]. Furthermore, we extended this study to the environment of the JET tokamak, by developing an MCNP model capable of describing the high-energy background of the JET tangential gamma ray spectrometer.

Due to the low probability of these events, several variance reduction techniques, such as Weight Windows, DXtran, F5 tallies are applied, and the results are compared. Finally, the simulation results are validated against the experimental results and the use of gamma-ray diagnostics for fusion power measurements is discussed in the context of the next-generation tokamaks, such as ITER, DEMO, BEST and SPARC.

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Design and Implementation of a Reinforcement Learning Based Plasma Shape Controller at DIII-D

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A novel Reinforcement Learning (RL) based controller for plasma shape and position control has been designed, implemented, and tested on the DIII-D. The RL controller was developed in NSFsim, a free boundary equilibrium and transport solver that has had its magnetic simulations validated to a high degree of accuracy with the DIII-D tokamak. The controller that the RL training produced, also known as the RL agent, was implemented in the DIII-D Plasma Control System (PCS) to run in real time and was configured to work with the DIII-D actuators. We tested the controller performance in a variety of plasma regimes and under transient events and were able to achieve the target shape with some errors.

Using our simulation tool, NSFsim, as a training environment, we trained an RL agent to be able to control the shape of the plasma to match a target shape using simulated magnetic sensors as input. This control task has been modelled as a Partially Observable Markov Decision Process (POMDP) and applied the Soft Actor-Critic algorithm to train the RL agent. We used an asymmetric Actor-Critic architecture with privileged information which stabilizes and accelerates the training process in the presence of uncertainties in the environment dynamics. A training pipeline was developed to streamline the training process, and now training a new controller can take less than 24 hours.

Implementing the RL agent into the DIII-D framework required both the development of real time PCS code and modifications to be made to NSFsim to model DIII-D actuators. The PCS code was written in C, had to pass strict speed tests on a 4 kHz loop, was written to be a determinant code as necessitated by real time programming, and was made to be flexible to different training coefficients (all RL Agents are the same size, 3-layer MLP with dimensions 132 X 256 X 18 with ReLU activation functions). To control plasma shape, the agent needed to control the F coils at DIII-D, but the PCS is limited to controlling the duty cycle of choppers connected to the F coils. The F coils current direction, chopper, power supply, and connection to ground or the VFI bus were subject to a configurable patch panel. These configurations and a simulated model of the choppers were implemented into NSFsim to enable a seamless transition for an RL Agent trained NSFsim to receive magnetic measurements and output chopper commands at DIII-D.

Two different RL agents were tested successfully on two different patch panels, across H mode and L mode, and across an H-L transition. The controllers managed to operate successfully despite the higher heating power being used in experiment than was used in training. The RL controllers performed very well in maintaining the plasma center, but the outer boundary did move significantly closer to the outer wall. We expect ongoing work to validate the transport models in NSFsim to improve the preciseness of the shape.

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Exploring the Application of Nuclear Safety Culture Framework to Fusion Energy: Focusing on ITER Project

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Thursday Posters 1, Lobdell (Building W20 Room 208), June 26, 2025, 10:30 AM - 12:00 PM

Currently, strategic partnerships between public and private sectors are increasing in the fusion energy development, aiming at bringing nuclear fusion-generated electricity to commercial feasibility in the near future (e.g. 2030s). Therefore, it is worthwhile to assess the nuclear safety culture in the fusion energy field as well.
International nuclear-related organizations, such as IAEA, INPO, WANO and US NRC, have developed frameworks to foster a positive nuclear safety culture, which have been applied to fission energy. In this study, the nuclear safety culture framework of the fission energy, developed by KHNP (Korea Hydro & Nuclear Power Co., Ltd.) and adapted to Korean cultural context from international nuclear-related organizations, is used to assess the nuclear safety culture of the domestic organization and companies in the ITER project for fusion energy. Data were collected through questionnaires to measure awareness of nuclear safety culture using a 5-point scale, and KANO Model questionnaires to assess the relative importance of six nuclear safety culture principles. Three research hypotheses were prescribed for the six principles of nuclear safety culture, and KANO Model was used for statistical test.
The following results were obtained in this study. First, safety training has a positive impact on the awareness of nuclear safety culture. Second, statistical tests indicate that safety training significantly affects nuclear safety performance. The relative importance of nuclear safety culture for fostering a positive safety culture is as the follow order: Leadership, Management System, Environment, Learning, Attitude, and Responsibility.
This study is the first trial to assess the relative importance of nuclear safety culture in the field of nuclear fusion energy development, as least within Korea domestic area. These findings indicate that the awareness of nuclear safety culture principles differs from those in fission energy.

Key words: Nuclear Safety Culture, Fusion Energy, ITER Project, KANO Model

Advances in fuel cycle modeling and analysis

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Achieving tritium self-sufficiency in fusion power plants requires advanced modeling to capture the dynamic phenomena of the fuel cycle accurately. This study addresses key challenges in fuel cycle analysis, including tritium trapping, integration of plasma physics parameters with fuel cycle dynamics, and uncertainties in tritium transport properties. We present results from novel models and tools integrated into a state-of-the-art fuel cycle framework: (1) physics-based modeling of irradiation-induced traps and tritium trapping dynamics in plasma-facing components and blanket structures, (2) explicit fueling and exhaust modeling for real-time estimation of tritium burn efficiency, (3) design-oriented models for outer fuel cycle components informed by tritium transport regimes and (4) uncertainty quantification using the TRIOMA Python package. Together, these advances enable a deeper and more comprehensive evaluation of tritium inventories and tritium self-sufficiency. Finally, this work offers actionable insights for the design of experiments to further improve fuel cycle modeling and support the development of tritium self-sufficient fusion power plants.

DEMO liquid breeder studies with IFMIF DONES neutron spectrum

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The eutectic PbLi is one of the candidates proposed as a tritium breeder for the EU DEMO. The Liquid Breeder Validation Module (LBVM) is designed to be irradiated in the International Fusion Materials Irradiation Facility DEMO Oriented Neutron Source (IFMIF DONES) to perform liquid breeder blanket experiments with the goal to demonstrate physical concepts. A group of capsules filled with the eutectic would receive an irradiation field similar to the one expected in the breeder zone of the future nuclear fusion DEMO reactor, producing tritium that is swept by a flow of helium and its concentration measured online. Some additional capsules will be lined with an antipermeation material in order to test how the coating behaves under this irradiation conditions. Finally, an independent set of capsules containing the eutectic as well, will host a bulk piece of ceramic material with the purpose to monitor the effect of irradiation in its electrical conductivity. The neutronic calculations to demonstrate the design soundness will be shown, as well as the conceptual mechanical design including He loops layout.
Another ongoing activity is the design of a Test Blanket Unit (TBU to be irradiated in IFMIF DONES, which includes a representative portion of the Water Cooled Lead Lithium (WCLL) blanket for DEMO. The key parameters determining the TBU neutronic behaviour have been calculated: tritium production, helium and hydrogen production, displacements per atom, nuclear heating and the ratios between them. An optimization of the geometry for the reduction of the transversal gradient (perpendicular of neutron propagation) is proposed and its improvement demonstrated. The results shown include a comparison of the values obtained for a representative part of the DEMO breeder zone and the TBU.

Dynamic Behavior of Radioactivity and Decay Heat During Pulsed Operation in Fusion Breeding Blankets

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Thursday Posters 1, Lobdell (Building W20 Room 208), June 26, 2025, 10:30 AM - 12:00 PM

Fusion power plants prioritize pulsed operation over continuous modes due to current plasma technology limitations. This operational mode induces fluctuations in neutron load on plasma-facing components (PFCs), directly affecting radioactivity and decay heat in critical components involving blankets and divertors. The radioactivity and decay heat in PFCs significantly impact tokamak safety, necessitating a thorough investigation of their transient behavior under pulsed conditions. This study investigates the transient behavior of specific activity and decay heat in a water-cooled ceramic pebble breeding blanket module under pulsed (2-hour plasma burn and 30-minute dwell) and continuous operations through numerical analysis. For the numerical analysis, a simplified box-shaped breeding blanket module with multiple layers was modeled. The first wall of module is made of tungsten, followed by alternating layers of a cooling channel composed of reduced activation ferritic/martensitic (RAFM) steel and a pebble bed made of Li_4SiO_4 and $Be_{12}Ti$ pebbles. The neutron load was selected to match that of the central outboard breeding blanket in a 2.2 GW fusion power tokamak, which undergoes the highest neutron flux. A coupled neutron-transport (MCNP) and inventory/source-term (FISPACT-II) model confirmed that the predicted shutdown radioactivity for a simplified breeding blanket module under continuous operation aligns with previous studies about radioactivation in K-DEMO. Under pulsed operation, the specific activity in the breeding blanket immediately after shutdown decreased by 34.8% compared to continuous operation. Analysis of key hazardous nuclides under regulatory oversight revealed reduced radioactivity in pulsed operation, demonstrating that specific activity is largely proportional to the total neutron irradiation time. While radioactivity is typically evaluated after shutdown, this study comprehensively examined the time-dependent behavior of decay heat, including its fluctuations during operation and after shutdown. Our results indicate that pulsed operation induces fluctuations in decay heat during plasma burn and dwell phases. After shutdown, pulsed operation generates up to 55% less decay heat compared to continuous operation. Analysis of dominant radionuclides in the tungsten layer, the RAFM steel coolant channel, and pebble bed identified the primary isotopes responsible for decay heat generation in each layer. These nuclide analyses confirmed that the fluctuation of decay heat is primarily driven by short-lived isotopes. Our findings provide critical insights for improving tokamak safety and advancing design optimization for pulsed operation.

Conceptual Design of Remote Vacuum Leak Detection System for Fusion Devices

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A high - vacuum environment provides a clean condition for high - temperature plasma fusion reactions and is of great significance for the high - parameter, safe and stable operation of magnetic confinement fusion devices. Vacuum leakage can affect the operation of the device, so it is necessary to quickly and accurately locate the leakage position. At present, the leakage detection of fusion devices mainly relies on manual on - site helium - spraying operations to determine the leakage location. In the future, when fusion devices operate with tritium and neutron involvement, it is impossible for personnel to work in the high - risk environment of the fusion device hall. Therefore, the research on a remote vacuum leak detection system is an urgent problem to be solved for the future operation of fusion devices. This paper proposes a data - driven rapid diagnostic method for leakage types, which is based on the historical operation data of the mass spectrometry and vacuum degree of the vacuum system and takes the machine learning algorithm as the core. Based on this method, through the way of helium - spraying inspection by a collaborative robot, the overall conceptual design of remote vacuum leak detection that can quickly and accurately locate the leakage position is realized.

Development and validation of nuclear data uncertainty quantification capabilities in OpenMC for fusion neutronics benchmarks

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Nuclear data uncertainty propagation is of paramount importance for nuclear fusion simulations to study the impact of TBR uncertainties on blanket design, heating uncertainties on blanket heat exchanger and cryogenic magnet design, and radiation damage uncertainties on thermomechanical analysis. The goal of this work is to illustrate the results of new capabilities implemented in the OpenMC Monte Carlo particle transport code for the uncertainty quantification (UQ) workflow applicable for fusion benchmarks. The first part of this work is the implementation of Fast Total Monte Carlo and Embedded Monte Carlo (EMC) methods in OpenMC. The novelty of this second method is the capability of performing only one large Monte Carlo simulation where each batch represents a new random sample, thereby embedding the propagation of uncertainties within a single calculation and reducing computational expenses. This work provides the first demonstration of EMC on fusion neutronics benchmarks. In addition to these methods, a new sampling capability is added to OpenMC to create random samples using covariance data. The goal of this work is to streamline everything in a single code to perform advanced neutronics analysis in an easy and direct way which could be really helpful for rapidly analysing the performance of fusion technology. Furthermore, the second part of this work is the implementation of new statistical tests and result analysis methods in OpenMC. These tests will provide users with the optional capability to better understand statistical fluctuations in their tallies while not degrading performance for cases where it is not necessary (e.g., in rapid design iterations where most tallies are large, volume-integrated quantities). Furthermore, these capabilities will help the user optimize the efficiency of their Monte Carlo simulation and provide more information about the statistical behaviour for the UQ workflow.

Structural Integrity Assessment of In-Vessel Components for ITER ECE Diagnostic

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The ITER Electron Cyclotron Emission (ECE) diagnostic measures the electron temperature profile, and the ECE radiated power from the plasma in the frequency range of 70 to 1000 GHz. The ECE diagnostic system includes two optical labyrinths located in the Diagnostic Shield Module 2 (DSM2) of Equatorial Port 9, that includes two in situ hot calibration sources, four mirrors, and two mirror-shutters.

As part of the preliminary design phase, a structural integrity assessment has been performed for the in-vessel mirrors, mirror-shutter, and calibration source support structure. In order to perform a complete structural assessment, the applied loads and boundary conditions are calculated in different analysis tools and imported into ANSYS for thermal and structural analysis. For example, nuclear heating loads are calculated in MCNP and electromagnetic loads are calculated in Maxwell. In some cases, these loads are calculated by other contributors (i.e. nuclear heating loads are provided by the Port Integrator). These loads are combined into different load combinations and analyzed in accordance with the ITER load specifications. The results from the analysis are checked against the failure criteria defined in ASME section VIII Division II of chapter 5.

This poster will present the combination of the different loads into the analysis and the structural integrity assessment as compared against the ASME criteria.

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Time Dependent Response of the ITER Diagnostic Residual Gas Analyzer Vacuum System

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The ITER Diagnostic Residual Gas Analyzer (DRGA) system measures exhaust gases in the divertor region. This measurement is critical for long pulse devices, such as ITER, and will provide important information for fuel-cycle processing and plasma heating [1]. The DRGA for the divertor analysis relies on an evacuated sampling pipe to transport the exhaust gases to the analysis station where spectrometers measure the exhaust gas composition [2]. At the sampling pipe entrance, an in-line orifice creates molecular flow conditions in the entire length of the sampling pipe. The current configuration of the sampling pipe will provide response times on the order of 1 second (low-amu gas species) in the RGA chamber of the divertor DRGA system on ITER. This determination is based on the mean pumping path and the most probable gas particle speed, which is not a time dependent model [3]. However, for plasma heating control and fuel-cycle processing, it is necessary to analyze the time dependent response resulting from changes in gas composition from injected gas and/or fueling pellets. Molflow+, which is a Test Particle Monte Carlo (TPMC) simulation code, is used to simulate the time dependent response of the DRGA from injections of hydrogen isotopes and 3He. In this work, the results of the time dependent response are shown from the sampling pipe orifice to the RGA and a plasma cell which is used for Optical Gas Analysis (OGA). The observed equilibration between core and divertor in ~1-s timescales is independently being studied by a time-dependent plasma edge modeling study [4]. This type of relationship between the plasma in the core and the divertor has been observed before and seems to equilibrate in ~1-s timescales [5]. The time dependent analysis will include both the RGA chamber and the inter-pump volume chamber where the plasma cell for the OGA is located. This study provides important design validation of the ITER DRGA time dependent response as it relates to fuel-cycle and plasma heating control.

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Comparison of Unstructured Mesh to Constructive Solid Geometry (CSG) and Sensitivity Study of Nuclear Data Libraries on Flux, Dose, and Heating in Fusion Neutronics Experimental Benchmarks

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SPARC is a compact, high-field tokamak using high-temperature superconducting magnets to achieve on axis magnetic field strengths of 12.2 tesla and targeting plasma performance sufficient to demonstrate commercial viability of fusion energy. A sensitivity study was performed to evaluate the performance of unstructured mesh (UM) geometry against constructive solid geometry (CSG) in MCNP to support the validation of neutronics models used in the design of SPARC. The sensitivity study was performed on two well-characterized fusion neutronics experimental benchmarks: The Fusion Neutronics Source (FNS) dogleg duct streaming benchmark and the Frascati Neutron Generator (FNG) ITER bulk shield benchmark. FNS is a facility at the Japan Atomic Energy Agency (JAEA) containing a 14-MeV deuterium-tritium based neutron generator for use in fusion-related experiments. The dogleg duct streaming experiment consists of a shielded duct with two 90-degree bends, with the neutron source at one end of the duct and detector points placed throughout the duct assembly. The FNG ITER bulk shield experiment uses a mock-up of the ITER inboard shield system, with alternating stainless steel and Perspex layers representing a blanket/vacuum vessel and a section of alternating stainless steel and copper representing toroidal field (TF) magnets. A 14-MeV deuterium-tritium based neutron source is placed on one end of the assembly and detector foils are placed through the mock-up. CSG models for each benchmark were provided by the Shielding Integral Benchmark Archive and Database (SINBAD) and were converted into CAD geometry using GEOUNED. CAD models were meshed and prepared for MCNP using Attila4MC. Neutron and gamma flux, dose, and heating MCNP results were compared between the two geometries for each benchmark. After comparing MCNP results in CSG and UM geometry, an additional sensitivity study was performed by repeating the previous MCNP runs with different nuclear and photoatomic data library inputs. Nuclear data libraries tested in this study include ENDF/B-VII.1, ENDF/B-VIII.0, and FENDL 3.2b, while photoatomic data consisted of the MCPLIB84 and eprdata14 libraries. MCNP calculations were compared against documented experimental results in different detector positions for each benchmark and repeated for each tested data library.

Rapid development cycles on compact fusion devices at Zap Energy

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Zap Energy is developing a system based on a sheared-flow-stabilized Z pinch (SFS-ZP) for fusion energy production. The Z-pinch device used at Zap is relatively low cost, modular and compact in size, with an overall diameter of less than 1 m and a length of less than 3 m. These characteristics make it unique amongst fusion energy devices and provide the opportunity for a short yet robust development cycle. The cost and compact size enable Zap to have several devices running concurrently. These devices are tailored to explore specific scientific questions, and dramatically reduce the configuration and maintenance iteration cycles.

Zap currently has four operational Z-pinch devices and is building a fifth. Two units, FuZE and FuZE-Q, are dedicated to advancing the physical understanding of plasma generation, compression and stabilization, with a mission to scale SFS-ZP systems to commercially relevant fusion energy gain. Two other units, SiMPL and Century, are used to develop technologies on the critical path to a fusion power plant. In particular, the Century platform integrates repetitive pulsed power and durable materials in a 100-kW, 0.3 Hz rep-rated, liquid metal plasma first wall device.

The primary purpose of the FuZE devices is to study the various phases of pinch generation: injection, ionization, acceleration, compression and sustainment. To date, Zap's devices have been primarily instrumented to characterize the pinch assembly area where fusion occurs. To improve understanding of the upstream accelerator region, Zap is now upgrading diagnostics access on one device and building a new made-for-purpose device dedicated to studying accelerator phenomena. This experimental capability will validate extended MHD models, improving simulation fidelity and further supporting future device designs.

These platforms are suited to quickly test key technical questions. For example, they are used to study various advanced electrode materials. They are also ideally suited to rapidly explore and optimize plasma-facing component geometries. Materials tests and configuration changes can be fielded within a few weeks to a few months depending on the complexity of the modification. This ability to rapidly iterate SFS-ZP devices accelerates the design-build-test-learn cycle and progress towards commercial fusion.

Cyclone Sage for Fusion: An Advanced AI Tool for Streamlined Analysis of Fusion Systems

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Integrating Artificial Intelligence (AI) into complex scientific workflows can address key challenges in achieving commercial fusion energy production at a commercial scale. Cyclone Sage is a chat-based AI assistant integrated within Cyclone™, a software platform designed to streamline Monte Carlo radiation transport workflows. Cyclone™ combines a modern CSG and CAD geometry viewer, 2D and 3D ray tracing, and advanced tools for visualising radiation data to provide a complete solution for developing and troubleshooting Monte Carlo models. Cyclone Sage enhances the Cyclone™ platform by enabling users to generate complete Monte Carlo input decks via natural language interactions. This method of model development demonstrates the significant value of AI in simplifying analysis workflows, and holds immense potential for application in the fusion industry.

This paper presents the adaptation of Cyclone Sage to interface with Paramak—a Python-based tool for generating parametric CAD models of fusion components and full tokamak reactors. Paramak enables users to define model geometries using adjustable parameters, facilitating the rapid design, analysis, and optimisation of complex fusion systems. The CAD models generated are imported directly into Cyclone™ for geometry inspection before analysis with Tempest—a new Monte Carlo radiation transport code developed by Orthrus Software. Tempest eliminates the need for CAD geometry conversions by performing simulations directly on native CAD geometries, significantly reducing calculation preparation time while maintaining high model fidelity. This workflow, combining Cyclone™, Sage, Paramak, and Tempest, demonstrates a streamlined approach to the analysis of advanced fusion systems, enabling calculations to be performed with both speed and accuracy.

Synthetic magnetic diagnostic integration of M3D-C1 simulations of SPARC

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This poster presents progress on coupling high-fidelity M3D-C1 simulations of plasma instabilities in the SPARC tokamak to ThinCurr (part of the OpenFusion toolkit) predictions of eddy currents to create realistic synthetic magnetic diagnostic signals. The framework simplifies the calculation of eddy currents from multiple resistive walls with realistic CAD-based geometries and arbitrary sensor placements. Phase- and amplitude- shifts are thus calculated directly. Such shifts can obscure diagnostic signals critical for accurate identification of magnetic perturbations in the plasma. Synthetic signal results are presented from simulation runs of self-consistent MHD modes, and for analytically defined modes. Details are presented on the conversion of 3D perturbed currents into closed filament representations necessary for the magnetic coupling calculations. Building on this approach, we describe progress towards: integration of higher-frequency magnetic probes for observability studies using simulations of high-frequency Alfvén eigenmodes, implementation of additional synthetic diagnostics, and integration with the data frameworks intended for SPARC operations. This final step unlocks the possibility of collaborating on data analysis tools for pre day-one operations of the tokamak. Following this, a longer term goal of this project is to produce labeled training datasets, using the simulated diagnostic signals from realistic simulations, for AI-enabled mode identification for tokamak operators.

Time-Resolved Neutron Emission Rate Measurements in the KSTAR Tokamak Using Micro Fission Chambers, Micro Helium-3 Detectors, LaBr3 scintillator, and Diamond-Based Detectors

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Measurements of time-dependent neutron emission rate is essential not only for understanding the fusion power and plasma behaviors in the KSTAR tokamak but also in other tokamaks. Time-resolved neutron emission rates on KSTAR were measured using micro fission chambers, micro helium-3 detectors, LaBr3 scintillator, and diamond-based neutron detectors [1]. This study presents the development and deployment of advanced diagnostic tools, including micro fission chambers, micro helium-3 detectors, LaBr3 scintillator, and diamond-based detectors, to further enhance the accuracy of measuring neutron emission rates under the challenging conditions of tokamak environments, including high-magnetic field, high-temperature and high-radiation conditions. The results indicate that these diagnostics collectively provide comprehensive temporal and spectral data, facilitating understandings of neutron production dynamics, confinement efficiency, and plasma behavior in KSTAR. The integration of these diagnostic systems demonstrates a robust approach to neutron flux measurement, paving the way for the development of improved neutron diagnostic strategies for future K-DEMO.

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Keywords: Micro Fission chambers, Micro He-3 counters, LaBr3 scintillator, and Diamond detectors, Neutron diagnostics, KSTAR, K-DEMO
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Structural Analysis and Design Optimisation of the ITER Port Mounted Bolometer Cameras

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ITER, the most important plasma fusion experimental facility under construction worldwide, is presently being built in southern France. The bolometry is one of the numerous plasma diagnostics installed within the device observing the plasma performance. The bolometer diagnostic provides the absolutely calibrated radiation emitted by the plasma, which is a part of the total energy balance. Bolometer cameras are positioned all over the plasma vacuum vessel including two upper ports and one equatorial port. These port mounted cameras vary in location and dimension, resulting in five different camera housing types. The bolometer camera located in the upper port, called upper camera, has been chosen as representative for consideration in the present paper due to the many challenges it has to face from design space and loads.

The development of the camera structural design is especially challenging because of the extreme environmental conditions within the vacuum vessel during plasma operation. Reliable measurements have to be assured while being subjected to high neutron fluxes as well as plasma radiation resulting in temperatures of the components exceeding 200 °C. In addition to the thermal loads, the bolometer camera housing is exposed to mechanical loads caused by electromagnetic forces (EM) during transient events of the plasma operation, called disruptions.

Due to the fact that the surrounding structure of the Diagnostic Shield Module within the upper and equatorial ports was already determined in advance, extensive design adaptation of all bolometer cameras was required. Extensive numerical analyses were carried out with the aim of achieving structural stability of the camera main body on the one hand and thermal requirements on the camera and sensors on the other. In a multiphysics analysis loop, several cycles were performed, starting with the electromagnetic followed by a thermomechanical and structural analysis. Additionally, the seismic and gravity loads and load scenarios like inductive and noninductive plasma radiation as well as fatigue cycles, need to be considered.

This paper gives an overview of such a comprehensive analysis on the example of bolometer camera mounted in upper port and presents the results achieved as well as the compromises taken.

Computational modelling and validation of Flow-induced vibration in WCLL-TBM design

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The ITER Water-Cooled Lithium-Lead Test Blanket Module (WCLL-TBM) features heat removal from molten lithium-lead (PbLi) through a set of immersed Double Wall Tubes (DWTs) carrying flowing pressurised water. Excessive vibration of the DWTs has the potential to lead to fatigue failure of the TBM over many cycles. The current study focuses on the experimental validation of a coupled CFD-FEM (Computational fluid dynamics - Finite element method) method used to analyse flow-induced vibration (FIV) of these DWTs. Previous studies have conducted FIV analysis on flow through double wall tubes of WCLL-TBM and assessed that the stress developed due to vibratory forces are well below the safety limit, but a full validation of the methodology utilised is necessary to ensure the accuracy of the simulation. The study emphasizes the validation of these models to bridge the gap between simulation and actual scenarios. A simplified experimental setup with geometry similar to that of DWT, along with digital image correlation (DIC) for measuring the vibratory displacements, was developed. The experimental setup constitutes a 180o U-bend Copper tube with ends fixed and the bend suspended like a cantilever. Water flows through the tube at a specific mass flow rate. The displacement of the pipe was captured with high spatial and temporal resolution using the DIC technique.
Flow through the U-bend was modelled using CFD and Large eddy simulation (LES) was used for turbulence modelling. As the simulation of the complete U-bend is highly expensive only the cantilever part of the test section is used for the fluid dynamics simulation. This assumption is justified by the fact that the turbulent fluctuation at regions away from the bend are smoothed out and, away from the cantilever end, the impact of pressure forces become lesser. Wall pressure fluctuation with a temporal resolution of 0.0004sec was extracted from CFD simulation to generate the frequency spectrum of pressure loading. The pressure loading on the pipe interior wall is then transferred to the structural FEM simulation to simulate the vibration response of the U-bend tube under study. The displacements derived from the one-way coupled CFD-FEM simulations were compared to the experimental measurements to validate the accuracy of the simulation. The spectral distribution of these measurements is used for comparison. A comparison is drawn between the experimental data and computational results and areas of agreement and discrepancies are identified and discussed. The results of the study help to determine the credibility of simulation techniques used for simulating flow-induced vibration in WCLL TBM components, and more generally advance techniques for multi-physics simulation to capture the complexities of TBM in-service conditions.