#11489

Reverse diffusion of deuterium in a lithium-lead blanket test loop

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The authors confirmed the reverse diffusion of dissolved deuterium on the experimental liquid lithium-lead test loop. The loop assumed the tritium breeding and extraction of the fusion reactor liquid blanket loop. In lieu of the bred tritium, deuterium is used as a solute.

The deuterium extraction efficiency is considered, a priori, as the concentration ratio of before-and-after extraction. The diffusion theory, meanwhile, tells that the half of dissolved solute diffuses in the opposite direction. The dispersion (dynamic diffusion) theory tells that the concentration in a flow is controlled by the solute diffusion and solvent convection. Therefore, the after-concentration, i. e., the residue deuterium after extraction, is to be merged with the reversely diffused deuterium if diffusion coefficient is relatively high. This phenomenon might affect the extraction efficiency, even though it is so far not reported.

Experimental verifications were performed on the Oroshhi-2 liquid metal test loop at National Institute for Fusion Science (NIFS), Toki Japan. The lithium-lead was circulated by the bore-through electro-magnetic-pump. The loop inner pipe diameter, length, and temperature is 42mm, 35m (approximately), and 350℃, respectively.

The reversely diffused deuterium is identical with the reverse-direction concentration distribution in the loop. The concentration distribution is observed in time-series by the loop-integrated method previously reported as VST. The released deuterium flux from the lithium-lead droplets, which is concentration proportional, is measured.

Two cases, 2 hours and 6 hours of deuterium dissolving, were examined with Reynolds number of 1.2×10^3, in laminar region. Both results accorded the diffusion and dispersion theory with an identical coefficient within 8% deviation. Obtained dispersion coefficient was 2.7×10^−2 m^2s^−1, which revealed seven orders of magnitude higher than that of the static diffusion. The enhancement almost obeyed the dispersion theory by Taylor which predicted the eight orders of magnitude. Ambiguities of the static diffusion coefficient and the loop piping details considered to contribute the difference.

On a practical fusion blanket system, the reverse diffusion affects not only the extraction efficiency but also the tritium inventory. Integration of the reverse diffusion blocking mechanism is recommended.

Recent progress in manufacturing the pellet launching system for JT-60SA

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The JT-60SA is a large superconducting tokamak intended to serve as a satellite experiment for the ITER project. As part of the enhancement programme within the Broader Approach framework, Fusion4Energy is purchasing a new centrifuge-based Pellet Launching System (PLS). Initially, the system is foreseen to be equipped with one pellet source to fuel the plasma core and one to pace the Edge Localized Modes (ELMs). Following the termination of the initial contract for manufacturing these two sources, a new staged approach to pellet source procurement has been implemented: one batch extruder for the short pulse phase of the JT-60SA and one continuous extruder for the long pulse phase for each purpose (four in total). The goal is to establish a supply chain within EU.
SENER SA has been awarded the contract to manufacture the centrifuge and the pellet diagnostic systems. During the factory acceptance test (FAT), issues relating to the lifetime of the bearings located in the vacuum were identified. The temperature of these bearings exceeds the acceptable range causing a minor vacuum leak in the ferrofluidic rotary feedthrough. The improved design aims to avoid placing additional loads on the bearings by using a flexible coupling between the rotor and the feedthrough axes. As this design change affects essential components, an additional factory acceptance test is required for the system in its final technical configuration. Any tests addressing the system's dynamic properties must be repeated.
Pellet sources and centrifuge are integrated to the PLS and controlled by a MasterPLC. This take place at IPP Garching pellet lab. A mock-up of this MasterPLC has been delivered to QST in order to enable the development of the interface to JT-60SA control system.
This contribution is intended to report the up-to-date status of this project.

Dedicated Thermal and Thermo-mechanical Analyses in support of DEMO WCLL BB design

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Within the framework of the EUROfusion Work Package Breeding Blanket (WPBB), several concepts of Breeding Blanket (BB) are considered for possible implementation in DEMO and are currently under development through different design phases. One of the concepts being investigated in EU is the Water-Cooled Lithium Lead Breeding Blanket (WCLL BB).
The present study focuses on the equatorial inboard slice IB4 from the steady-state thermal and thermo-mechanical point of view for Normal Operating (NO) and In-box LOCA Over-Pressure (OP) conditions. The thermo-mechanical evaluation uses the thermal field imported from dedicated thermal calculations and assesses RCC-MRx rules (focused on type P damages). In 2025, based on previous studies, the DEMO baseline was updated and replaced by the so-called Low Aspect Ratio (LAR) DEMO baseline. The IB4 study with this new baseline is presented in this paper.
Regarding the previous DEMO baseline applied on the IB4, first thermo-mechanical calculations revealed that the criteria associated to negligible creep and significant irradiation rules were exceeded for NO conditions.
Finally, the thermal and thermomechanical results corresponding to LAR DEMO highlighted significant improvements. Temperature fields have significantly decreased, which is beneficial in terms of creep aspects. Indeed, the maximum temperature to stay in negligible creep whatever holding time (375°C) is only exceeded on the front part of the slice. The thermomechanical stresses have decreased and the criteria not meet previously are now satisfied.

Preliminary Thermal and Mechanical Assessment of a Modular Monoblock HCPB Breeding Blanket for EU DEMO

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The Helium-Cooled Pebble Bed (HCPB) Breeding Blanket (BB) concept is one of the breeding blanket candidates for the European fusion demonstration power plant (DEMO) within the EUROfusion framework. The HCPB BB is expected to fulfill several mission-critical reactor functions, however, its current design and fabrication present several challenges. These challenges have motivated to evolve the architecture towards a significantly simplified configuration, referred to as the Monoblock HCPB.

In the HCPB Monoblock design, the First Wall (FW) and the Breeder Zone (BZ) structures are manufactured as a single solid block. This innovative design, currently under development at KIT, aims to overcome the limitations associated with the existing BB reference design and its fabrication & assembly procedures. By incorporating a compatible Fuel Breeder Pin Assembly (FBPA), the FW region can be cooled effectively without the need for a dedicated actively cooled FW component. In addition, the monoblock configuration supports modularization of the FBPA, in which the neutron multiplier is contained within canisters. Another novelty of this architecture is the replacement of TiBe12 blocks with more cost-effective Pb Canisters acting as neutron multipliers.

The present work discusses the rationale behind this design evolution of the HCPB concept and evaluates the preliminary nuclear, thermal-hydraulic and thermo-mechanical performance of the monoblock configuration under normal operating (NO) conditions. Preliminary neutronic analysis have resulted in a Tritium Breeding Ratio (TBR) of ~ 1.18 for the presented monoblock design. Initial thermal-hydraulic estimations predict temperatures close to 665°C at the Pb canister walls; however, since, these canisters do not form part of the pressure boundary and the operating pressures of the coolant and purge gas are equal, no structural failure is anticipated. Preliminary thermo-mechanical calculations indicate that the design can withstand under NO conditions, satisfying Level A criteria of the RCC-MRx: N2RX code.

Analytical Modelling and Numerical Investigation of MHD Pressure Drop in Rectangular Ducts under Spatially Varying Magnetic Fields

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

In Dual Coolant Lead Lithium (DCLL) breeding blanket concepts, a Lead-Lithium alloy flows through electrically conducting ducts exposed to intense magnetic fields, resulting in significant magnetohydrodynamic (MHD) pressure losses and critical pumping power requirements. Indeed, due to the motion of the electrically conducting fluid across the magnetic field, electric currents are induced within the flow, giving rise to Lorentz forces that oppose the fluid flow and strongly modify the velocity distribution. In this framework, predictive analytical tools could speed up the preliminary design assessment phases for the development of future breeding blankets.

The present work is focused on the analytical evaluation of the MHD pressure drop on a Lead-Lithium flow in a rectangular duct under different non-uniform magnetic field conditions. The analytical approach, already validated on spatially uniform magnetic fields, is based on an equivalent electrical circuit describing the current closure through the fluid core, boundary layers and conducting walls, where the electrical resistances also depend on MHD parameters. These analytical predictions are compared with three-dimensional CFD simulations carried out using a commercial finite element software where the effects of magnetic field gradients on the flow structure, electric current distribution and pressure losses are evaluated. This comparison shows a similar trend in terms of pressure drop scaling with magnetic field intensity.

#12820

Compositional impact on thermal properties of Li2BeSiO4

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

To realize commercial fusion reactors, it is necessary to ensure safety of the reactor. One of the concerns is in-box loss-of-coolant accident (LOCA), which could be caused by breakage of degraded cooling pipes. If a LOCA happens with pressurized-water cooling system, the oxidation of beryllium by water produces hydrogen gas and potentially leads to hydrogen explosion. Since the risk cannot be eliminated completely, strategies to enhance LOCA tolerance are required. Recently, Li2BeSiO4 has been proposed as a new breeder material that can enhance the safety level of fusion reactors since this material has excellent resistance against LOCA conditions. Furthermore, Li2BeSiO4 is expected to have higher thermal conductivity than that of current promising candidates for breeder materials. However, the thermal properties of Li2BeSiO4 have not been experimentally investigated. Also, the effect of compositional changes under neutron irradiation on the thermal properties has been unclear.

In this study, we synthesized Li2BeSiO4 specimens with and without reduced Li2O content to simulate the compositional changes due to Li burn-up. The starting materials were mixtures of LiOH·H2O, BeO, and SiO2 powders with a molar ratio of 2(1-x):1:1, where x was set to 0.00, 0.05, 0.10 and 0.20.. Then, we performed thermal conductivity measurements on these specimens using the laser flash analysis (LFA), simultaneous thermogravimetry–differential scanning calorimetry (TG–DSC), and He pycnometry. Also, X-ray diffraction (XRD) and scanning electron microscopy (SEM) were carried out to analyze the phase composition and microstructure. As a result, it was found that Li2BeSiO4 had a higher thermal conductivity of 5.1 W m^{-1} K^{-1} at room temperature than those of current promising materials such as Li2TiO3 and Li4SiO4. Also, it was observed that while the skeletal density decreased with reduction of Li2O content, the thermal conductivity increased. We considered that this was because a BeO phase, which has a higher thermal conductivity than 225 W m^{-1} K^{-1}, was formed with a reduction in Li2O content. Neutron irradiation introduces phase changes and irradiation defects in materials, likely decreasing the thermal conductivity. Our study suggests that Li burn-up alone increases the thermal conductivity of Li2BeSiO4. It is possible that Li burn-up can partially suppress the reduction in the thermal conductivity of Li2BeSiO4 under neutron irradiation conditions.

Experimental Characterization of a Gas–Liquid Contactor Column for Tritium Extraction from Flowing Lead–Lithium under WCLL TBS Operating Conditions

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The Gas-Liquid Contactor is the reference technology for tritium extraction from PbLi in the WCLL TBS (Water Cooled Lithium-Lead Test Blanket System) of the ITER reactor. However, experimental data are limited to the experiments performed at the CEA MELODIE loop in the late 1990s.

In this work, a 1:1 scale GLC mock-up was experimentally characterized at the TRIEX-II facility of ENEA Brasimone Research Centre under WCLL TBS-relevant operating conditions, covering temperatures between 400 and 460 °C, PbLi mass flow rates between 0.2 and 1.4 kg/s, and inlet hydrogen partial pressures in the range 93–290 Pa. These conditions are also partially relevant for the Breeding Blanket conditions of EU DEMO. Hydrogen was used as a surrogate for tritium in PbLi and helium was employed as the stripping gas in the column.

The GLC extraction efficiency was evaluated by combining measurements of hydrogen partial pressure in PbLi, obtained using Hydrogen Permeation Sensors installed upstream and downstream of the column, with gas-phase measurements performed by a quadrupole mass spectrometer at the helium inlet and outlet. The experimental results show an increase in extraction efficiency with temperature and a clear sensitivity to the liquid-to-gas flow rate ratio, while a weaker dependence on the inlet hydrogen partial pressure is observed. Overall, the TRIEX-II experimental campaigns demonstrate that packed-column Gas–Liquid Contactors can reproducibly achieve single-stage hydrogen extraction efficiencies on the order of 10–20% in PbLi. An uncertainty analysis of the experimental data is presented to quantify the error associated with the efficiency calculation.

To support scale-up efforts, a model was developed to extract mass transfer coefficients from the experimental data. The model integrates Sieverts' law, interfacial mass balances across the gas-liquid metal interfaces, and conventional stripper column modelling assumptions. The obtained mass transfer coefficients fall within the expected range of conventional contactors. Furthermore, there is a positive correlation between the mass transfer coefficients and the liquid metal flow rate, as expected from mass transfer theory. Additional analyses of the results are ongoing.

#12851

Development of analytical techniques for real-time tritium accountancy around the fusion fuel cycle using the HERA experiment

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The HERA experiment (Hydrogen Experiment for Real-time Analysis) is looking at testing analytical methods for real-time analysis of tritium in gas streams. The project is a glovebox experiment installed in the Active Gas Handling System (AGHS) building on the Culham campus, in Oxfordshire, UK.

The glovebox has been entirely designed by UKAEA engineers and has been built in the facility; completion of the internal mechanical build was due end of February 2026. Commissioning, started in March 2026, first tested the glovebox itself for safety purposes followed by testing and calibration of the analytical instruments for hydrogen, deuterium and tritium. Commissioning is split into inactive and active campaigns, with the inactive campaign planned to start in March 2026 and the active one in November 2026.

HERA is testing 2 analytical instruments: a mass spectrometer (MS) and a Raman spectrometer. The Mass spectrometer is a single quadrupole MS from HIDEN configurated for continuous analysis with ultra-high resolution to enable analysis of all hydrogen isotopologues. The Raman spectrometer is custom made and comprises of 3 gas cells with different length of fibre to test size of gas cell versus accuracy and response rate.

The HERA glovebox is capable of generating its own gas calibration standards thanks to calibrated vessels and PVTc calculations. The instruments are able to quantify H2, D2 and T2 from limit of detection up to 100% gas streams.

A Palladium catalyst is also installed in the glovebox to allow for isotopic exchange and semi- quantification of HD, HT and DT.

We will present initial commissioning results, obtained recently, comprising of instruments testing with inactive gases: hydrogen and deuterium, to be followed with tritium testing later this year.

#12853

Water Detritiation System for the JA DEMO.

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

In the tritium fuel cycle, the Water Detritiation System (WDS) plays a central role in recovering tritium from aqueous streams and returning it to the fuel cycle. Among various water detritiation technique, JA DEMO adopts a Catalytic Exchange Combined Electrolysis (CECE) as the primary WDS concept. It processes aqueous streams originating from multiple sources. One major source is tritium permeation into the primary cooling water. Another source is tritium-contaminated water generated by the Air Detritiation System (ADS) and the Hot-cell Detritiation System (HDS), both of which employ molecular sieve beds in nominal and scrubber columns in an accident to remove tritium from room air. A key feature of the WDS is its close integration with the Isotope Separation System (ISS), which supplies hydrogen isotopes required for the catalytic exchange process in the liquid-phase catalytic exchange (LPCE) column. Therefore, the design of the WDS must consider the throughput and isotope composition of both water and hydrogen streams. Furthermore, safety considerations related to hydrogen explosion risk due to oxygen gas produced by the electrolysis cell, as well as the release of tritium-containing hydrogen gas and water vapor from the LPCE column, must be addressed. Due to the limited availability of large-scale experimental data for systems handling highly contaminated tritiated water, the WDS design relies heavily on numerical simulation. Based on a simulation code validated against pilot-scale CECE operation at the Fugen Advanced Thermal Reactor, this work aims to present a conceptual configuration of the WDS under continuous steady-state operation by conducting a parametric study to explore feasible design windows that satisfy both fuel recovery requirements and safety constraints. With reference to previous WDS design studies, system specifications—such as processing flow rate, hydrogen isotope composition, detritiation factor, and regulatory limits for tritium-containing hydrogen gas and water vapor—are defined. Based on these specifications, the design study evaluates tritium recovery and release, tritium inventory in liquid hold-up, geometric characteristics—particularly the height of the LPCE column—and compatibility with the ISS, while considering the introduction of a water distillation WD upstream of the CECE process as an option to reduce system load and tritium inventory.

#12877

Pressure drop and flow distribution in a mock-up of the ITER WCLL TBM: numerical investigation of 3D magnetohydrodynamic flows

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The Water-Cooled Lithium Lead (WCLL) breeding blanket is one of the four concepts foreseen as a Test Blanket Module (TBM) for ITER. In this concept, the heat generated by neutron interactions is removed by water-cooled pipes immersed in a liquid eutectic lead–lithium (PbLi) alloy, which serves simultaneously as tritium breeder and heat transfer medium. A comprehensive understanding of the complex three-dimensional magnetohydrodynamic (MHD) phenomena governing PbLi flow is essential to assess the feasibility and performance of liquid metal blankets.

Experimental studies with liquid sodium–potassium alloy (NaK) were performed in a 1:2.5 scaled mock-up of the ITER WCLL TBM at the MEKKA laboratory (KIT). Electrical potential measurements along the Hartmann walls of the eight breeding units (BUs) revealed a strongly non-uniform flow distribution. Such non-uniform PbLi flow and residence times affect tritium transport, leading to localized accumulation or depletion, reduced breeding performance, and potential safety concerns. Quantitative evaluation of individual BU flow rates was not possible due to the non-negligible Hartmann wall thickness; Therefore, numerical simulations are required to obtain detailed flow and velocity distributions.

In this study, nondimensional results from full numerical MHD simulations are validated against experimental data. The simulations provide additional insight into pressure-drop mechanisms beyond experimental observations, resolving detailed flow features in the manifolds, in particular the sharp pressure drops at local contractions and expansions caused by stiffening plates penetrating the manifolds. Consequently, the highly resolved numerical results provide a more accurate estimation of the breeding-unit pressure drop.

Feasibility Study of a multi-stage cryopump for the DIR of a compact tokamak

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Particle exhaust management is a key challenge in the design of future fusion power plants. The primary pumping system must operate across a wide range of pressures and throughputs, while fulfilling safety requirements including limits on hydrogen concentration and tritium inventory. Cryopumps are a mature technology for this role. Moreover, when implemented in a multi-stage configuration, they can additionally provide gas separation enabling a direct internal recycling (DIR) loop. In this work, we investigate the feasibility of the multi-stage cryopump technology as a primary pumping system for compact tokamaks and assess its performance over a broad range of operating conditions, explicitly quantifying the design margin needed to accommodate high puffing rates. Plasma exhaust conditions and gas puffing requirements are systematically scanned to identify pump configurations that remains effective across different operating scenarios, while respecting the geometric and integration constraints of compact reactors and other relevant requirements. Impurity effects are treated consistently at two levels. On the pumping side, we account for species-dependent pumping speeds and separation sharpness between hydrogenic species and impurities. On the plasma side, we model impurity content from the DIR return stream to evaluate the impact of residual impurity species (notably H and Ne) on fuel dilution, radiative losses, and optimal DIR fraction. By explicitly coupling pump performance with edge plasma and fuel cycle constraints, this analysis moves beyond prescribed fixed boundary conditions at the plasma–pump interface and instead investigate dynamic interactions between the pumping system and the plasma. This integrated approach supports the identification of feasible cryopump configurations capable of realizing a DIR loop in the fuel cycle architecture and quantifies how this coupling shape DIR performance in the presence of impurities.

Comparative Analysis of Numerical Efficiency and Scalability of Surrogate Model Types for Integrated Fusion Fuel Cycle Optimization

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The fusion fuel cycle is an intricately interconnected system comprising sub-processes like ISS, WDS, and TEP. For achieving tritium self-sufficiency and economic viability, integrated system-level optimization is imperative. However, high-fidelity physics-based models are often unsuitable for direct integration into optimization loops due to high computational costs. This study proposes a numerical optimization strategy using surrogate models within the IDAES framework to overcome these challenges.

Acknowledging that the development of full-scale high-fidelity models for some sub-systems is still in progress, this research was conducted as a Proof of Concept using general chemical plant datasets and limited process simulation data. Consequently, the focus of this presentation lies in evaluating the numerical suitability of various surrogate architectures—specifically how their mathematical structures (e.g., activation functions and complexity) interact with optimization solvers—rather than achieving a full representation of the fusion fuel cycle's physical behavior.

Diverse surrogate types, including ANN (ReLU, Softplus, Tanh), Symbolic Regression (ALAMO), and Kriging(PySMO), were analyzed through two primary case studies: (1) Scalability analysis, assessing the impact of increasing the number of units (1 to 8) on computational load; and (2) Complexity analysis, identifying the optimal balance between model fidelity and convergence stability. This study provides essential strategic guidelines for the immediate and seamless integration of detailed sub-system models once their development is finalized.

Preliminary Design Development of Water-Cooled Ceramic Breeder Test Blanket Module

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

To further the development of early-stage fusion power generation, Japan is advancing the Water-Cooled Ceramic Breeder (WCCB) concept as the primary candidate for the DEMO breeding blanket. The WCCB Test Blanket Module (TBM), which serves as a prototype for the DEMO breeding blanket, is scheduled for installation in equatorial port No. 18 of ITER. The main objectives of this initiative are to demonstrate and collect essential data regarding thermal and tritium extraction performance, which will aid in the development of the future DEMO breeding blanket. As part of the WCCB TBM development process, a preliminary design review is planned for 2026, with comprehensive investigations and evaluations addressing structural integrity and nuclear performance to be outlined in this study. The RCC-MRx code was utilized to assess structural integrity through single load and load combination numerical analyses. Neutronics analyses were conducted to examine nuclear heating distribution and the local tritium breeding ratio.
Based on neutronics analyses, the local TBR of a single submodule is found to be 1.21. The nuclear heating distribution of the WCCB TBM set has been quantitatively investigated and will be used for further thermal and thermo-mechanical analysis. It was also revealed that the supporting structure requires the implantation of cooling channels to mitigate nuclear heating. A structural integrity assessment indicated that the integrity of the submodule can be effectively evaluated. However, the supporting and pin structures, which play a crucial role in the mechanical connection between the TBM and the shield, exhibited stress magnitudes that exceed the RCC-MRx criteria induced by Maxwell forces. This suggests that further structural optimization is essential. The design of the cooling system demonstrates adequate cooling capacity to manage both surface heating and nuclear heating during ITER operations.
In summary, our current design and achievements for the WCCB TBM are well-positioned to support the preliminary design development. More detailed investigations and evaluations will be presented in subsequent discussions.

Improvement of electrochemical isotope separation methodology for future large-scale lithium-6 production

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Reliable supplies of lithium-6 are essential for sustaining tritium production in future thermonuclear fusion reactors, where breeding blankets rely on enriched lithium to generate tritium during operation. Natural lithium (~7.5% 6Li) must be enriched to 70 – 90% via separation processes to be usable, and the historical COLEX amalgam process, although effective, has largely been discontinued due to severe environmental concerns and mercury waste production. However, most alternative lithium isotope separation technologies remain at relatively low technology readiness levels and have yet to demonstrate industrial-scale viability. With several fusion reactor pilots approaching operation, the demand for enriched lithium-6 is projected to reach hundreds of kilograms to several tons per installation, highlighting the urgent need for scalable and sustainable separation technologies.

Electrochemical isotope separation offers a promising pathway as it can be easily scaled up, powered by low-carbon renewable energy sources, and implemented in continuous, as opposed to batch, processes. Among the different systems studied, electrodialysis shows particularly great promise. In a typical system, the 6Li is preferentially separated from a feed stream via the migration of the lithium ions through a selective membrane under an electric field. However, progress in this emerging field is hindered by inconsistent methodologies and reporting practices that complicate comparisons across studies and obscure the underlying separation mechanisms. In an effort to improve on that, we demonstrate that the measured separation coefficient strongly depends on the catholyte sampling method. In other words, for a set separation system, several separation coefficients, an important performance metric, can be measured based on the sampling approach, leading to large inaccuracies. Conventional partial sampling yields a cumulative separation coefficient that reflects overall enrichment but may give distorted results depending on sampling volume and frequency. We introduce a catholyte renewal protocol enabling the estimation of an instantaneous separation coefficient at a given time, which opens the way for better mechanistic interpretation and enables comparison with theoretical models.

Because industrial deployment requires both high selectivity and high throughput, we further propose a simple figure of merit combining separation coefficient and lithium ionic flux. This metric provides a realistic basis for identifying optimal operating conditions and materials, and prevents the reporting of artificially high performance metrics. Establishing unified measurement protocols and performance metrics will accelerate the development of industrially viable lithium-6 enrichment technologies for future fusion energy systems.

A blanket functional materials irradiation module in IFMIF-DONES for Helium Cooled Pebble Bed breeding blanket

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Within the EUROfusion DEMO programme, the Helium Cooled Pebble Bed Breeding Blanket (HCPB BB) is one of the candidate driver blankets for European DEMO. The HCPB BB utilizes pressurized helium as a coolant, lithium-containing ceramic pebbles as the tritium breeder, titanium beryllide blocks as the neutron multiplier, and Eurofer97 as the structural material. The current HCPB BB design is based on a fuel-breeder pin concept, where the lithium ceramic pebble bed is enclosed within a Eurofer-steel pin, with the titanium beryllide block positioned outside. The tritium breeder and neutron multiplier materials are collectively referred to as the functional materials of the HCPB BB. Extensive design and experimental studies have been conducted on these materials, focusing on single-effect phenomena such as thermal cycling behavior, tritium release properties, and crush characteristics. To investigate multiple-effect phenomena under DEMO-relevant conditions, a fusion-like neutron source with sufficient irradiation volume, a relevant neutron spectrum, and an appropriate flux is required. The IFMIF-DONES (International Fusion Materials Irradiation Facility – DEMO Oriented NEutron Source) provides neutrons in a standard footprint of 20x5 cm^2. In the second row behind the High Flux Test Module, and assisted by neutron reflectors, the flux is spread out sufficiently to extend over a HCPB breeder zone unit cell and still exceed 1e13 $n/cm^2/s$, suitable for studying multiple-effect phenomena in the breeder zone of the HCPB BB using a mockup that replicates DEMO conditions to an extent that can qualify the component up to a nuclear technology readiness level of 5-6. For this purpose, a dedicated mockup called BLUME (BLanket fUnctional Materials irradiation modulE) has been developed. This work presents the system requirements, test objectives, preliminary design, and analyses of BLUME.

#12944

The Dynamics Behind Steady State Fusion: Modelling Challenges and Progress Towards Integrated Solutions for STEP

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Steady-state fusion powerplants offer plant-level advantages over pulsed operation, but fuel-cycle boundary conditions remain dynamic. Maintaining a stable plasma during ramp up, controlled exhaust operation, and batch or semi-batch processing steps across the fuel cycle introduce time-varying flow and throughput demands. These transients can propagate through coupled recycle loops and can affect tritium inventory, buffering requirements between systems, and introduce challenges to supplying a constant composition-controlled fuel of high purity.

This work outlines the challenges in capturing these effects and reports progress toward an integrated, system-level fuel -cycle model for STEP. The approach uses reduced order unit operation models accounting for holdup inventory, residence-time delays, whilst ensuring material balances within units and across the flowsheet. The integrated model supports steady-state studies of fuel cycle architecture as well as time dependent assessment of start-up behaviour, disturbance propagation, and off-normal scenarios.

The work focuses on two analysis themes that are not well captured by component-only or purely steady-state approaches: the effect of cyclic torus vacuum pumping and associated transients on inner fuel-cycle performance and interface requirements; and longer-term fuel self-sufficiency and inventory studies, with emphasis on key plant interfaces that shape tritium availability and overall inventory.

#12947

Design and Prototyping of a Centrifugal Pellet Injector for the Fusion Energy Industry

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

A new, commercially focused pellet injection system is being designed at General Atomics (GA) to allow high repetition rate and high purity fuel injection into both experimental and commercial fusion energy reactors.

While previous pellet injection systems have successfully demonstrated launching pellets with diameters and lengths of up to several millimeters, the GA pellet fuel injection system aims to launch pellets up to 10 mm in diameter and length with a linear speed of up to 1000 m/s. This centrifugal acceleration of the fuel pellet requires the rotational arm, referred to as an “arbor blade” to spin at rotational speeds up to 10,000 rotations per minute (rpm). This contribution presents the design approach to this engineering challenge, including the results of an iterative and comprehensive finite element arbor analysis and integration of the arbor blade into the larger pellet fuel injection system. Design solutions and material choices are identified to meet system requirements: mechanical stresses are minimized by careful shaping of the arbor blade, and high-grade titanium is selected for the arbor blade to allow 10,000 rpm operation without exceeding structural fatigue limits. The continuing development of a prototype pellet injector system is discussed.

Funding Acknowledgement: This work is funded by General Atomics corporate funding and the INFUSE program – A DOE SC FES public-private partnership – under CRADA number NFE-24-10580 between Oak Ridge National Laboratory and General Atomics.

Current R&Ds on advanced breeding functional materials for JA DEMO

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Advanced breeding functional materials, neutron multiplying materials and tritium breeding materials for JA DEMO applications are essential to achieve a high tritium breeding ratio (TBR) and to secure highly-safe systems in the JA DEMO blanket.
The latest blanket concept for JA DEMO has been changed from loading a mixture of beryllide pebbles and Li2TiO3 pebbles in honeycomb structures to loading beryllide blocks and Li2TiO3 pebbles in cylindrical structures.
QST has enthusiastically dedicated to establish the database of beryllides (Be12Ti) block including the fundamental properties, mechanical properties etc. In addition, we have succeeded in producing the Li2TiO3 pebbles as well as advanced pebbles, Li added Li2TiO3 and Li added Li2TiO3 with Li2ZrO3 solid solution (LTZO) using an emulsion process.
In terms of R&Ds on characterizations, we have carried out a wide range of R&D activities based on the requirements of blanket design, thermal properties, mechanical properties, chemical stability with atmospheres and other materials, irradiation properties and so on.
In this presentation, the current R&Ds on advanced functional materials in JA DEMO activities and also the first results of neutron irradiation properties for these functional materials will be presented. In addition, the recycling process and scaleup approaches of these materials established by the QST will be addressed.

#13004

Breeding Blanket Neutronic Studies of ARC-like reactor

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

High field compact fusion reactors, such as ARC-class devices, impose stringent neutronic requirements on breeding blanket (BB) systems, demanding robust tritium self sufficiency, effective shielding of superconducting magnets and controlled material degradation under intense neutron fluxes. This study performs a comprehensive neutronic parametric assessment of multiple BB concepts using different cooling fluids, breeders and tritium carriers. The scope is to identify the neutronic trade-offs between tritium breeding performance, shielding effectiveness, and structural damage limits among the different BB concepts.
The work is structured in two phases. In Phase I, simplified homogeneous and semi homogeneous models are used to systematically explore the design space, varying key parameters such as geometry, material composition, 6Li enrichment, coolant type, and neutron source characteristics. Resulting outputs, including power density, tritium breeding ratio (TBR), nuclear heating on structural and breeder material, neutron flux spectra, displacements per atom (dpa), helium production, and shielding performance are extracted and added into a structured database.
In Phase II the generated dataset is used for cross-concept comparison, data filtering and normalization and to construct surrogate Response Surface Models (RSMs) based on regression techniques. These models enable sensitivity analyses, uncertainty quantification, and rapid performance prediction across the explored space. Performance envelopes defined using representative engineering targets are applied to screen viable concepts, A weighted scoring methodology is then used to identify promising regions of the design space and highlight dominant neutronic mechanisms governing BB performance. The resulting database and validated RSM framework provide a foundation for accelerated design of BBs for high field fusion reactors.

Overview of current tritium-based R&D efforts for fuel cycle instrumentation and storage technologies at TLK

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

For the realisation of the fuel cycle of a Fusion Power Plant (FPP), the Tritium Laboratory Karlsruhe (TLK) and Kyoto Fusioneering Europe (KFEU) are jointly pursuing two key technology development activities. In the first project, a detailed survey of current tritium detection techniques including their metrological characteristics, e.g., sensitivity and accuracy, is combined with a simulation workflow in order to assess the potential mass balance areas and the combined uncertainties of continuous measurements. For this activity, consideration of the different strengths and drawbacks of the various detection methods is performed to select the appropriate device for the relevant fuel cycle system. In a second research effort, a new testbed facility is in the setup process at TLK. Aiming at assessing the thermodynamic and kinetic properties of hydrogen absorption for different metal hydride storage materials, such as ZrCo alloys, with high accuracy and up to 3 grams of tritium, this task can help mitigate challenges in both, storage and non-proliferation (replacement of uranium beds). Equally vital is the investigation into the disproportionation and regeneration of these materials.

This poster provides an overview of the status of both the testbed facility for metal hydride storage materials and the fuel cycle instrumentation selection for tritium activity monitoring.

Impact of humidity contamination on the Performance of Palladium-Based Membranes

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The composition of hydrogen isotopes, primarily Tritium (T) and Deuterium (D), constitutes the fundamental fuel mixture in fusion reactors. However, only a very small fraction of the injected isotopes is actually consumed in the plasma chamber. Therefore, the recovery, purification and separation of hydrogen isotopes are crucial steps for efficient fuel management, safety and overall process performance. The fuel cycle encompasses the systems that receive plasma exhaust and other tritiated streams originated from several sources (i.e., blanket extraction system, ventilation lines, purge gases, etc.) for detritiation and recirculation purposes by means of interconnected processes and technologies.

Among the available technologies, Palladium (Pd)-based membranes represent a safe and energy-efficient solution for hydrogen isotope purification in nuclear fusion applications due to their high selectivity and permeability. Their use has therefore become a reference approach whenever impurities must be removed from hydrogen isotope rich gas streams. This is the case, for example, in the Exhaust Processing System, where impurities must be separated from the plasma exhaust, and in the Water Detritiation System, where humidity must be removed prior to sending the hydrogen isotope stream to the Isotope Separation System.

However, membrane performance may gradually deteriorate because of contaminant-induced effects, including competitive adsorption, surface poisoning, and structural degradation.

This work investigates the influence of selected impurities on membrane surface phenomena and bulk diffusion transport mechanisms. Particular attention is given to humidity-induced effects, as well as the impact of reactive species on hydrogen permeation, flux decline, and the reversibility of poisoning processes. Experimental observations are supported by transport analysis and a mechanistic interpretation of adsorption–diffusion interactions.

The results provide insight into performance degradation mechanisms and recovery strategies, contributing to the optimization and long-term reliability of membrane-assisted hydrogen isotope separation systems for future fusion reactors.

Keywords: fuel cycle, hydrogen permeation, fusion technologies

Experimental assessment of a Pd-Ag double-stage permeation system for high efficiency Hydrogen recovery

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The ENEA laboratory in Frascati has a long experience with Pd-Ag permeator devices. Activities have included the development of Pd-Ag membranes, the design and manufacturing of prototype systems with innovative mechanical solutions, and the construction of facilities for testing single and multi-tube configurations. Inside the fuel cycle, such Pd-Ag permeators found their principal application in the exhaust process system, where the unburnt D-T fuel needs to be separated from impurities in a fast and continuous manner. In Pd-Ag membranes, the separation occurs thanks to the high permeability and infinite selectivity toward hydrogen isotopes, and the permeation flux is proportional to the difference between the hydrogen partial pressures, between the two membrane sides, following a power-law dependence with exponent n between 0.5 and 1. This relation highlights the importance of the interface among the membrane and the pumping units as, inside the fuel cycle, an important fraction (between 20 and 100%) of the tokamak exhaust must be processed inside the membrane unit. This work presents the results of an experimental campaign performed in a Pd-Ag double-stage permeation system realized by connecting in series two set-up named MeSMeR and HyPe. MeSMeR is a multi-tube facility hosting 10 Pd-Ag membranes in parallel with a total membrane area of 0.1439 m2, whose shell side is connected to a scroll pump having an ultimate pressure of 3 Pa. HyPe is a single-tube module with an area of 0.0159 m2 connected to a turbopump with an ultimate pressure of 1x10-8 Pa. The HyPe module has been designed and constructed with an optimized heating unit and a reduced conductance between the shell side and the pumping system. The experimental campaign consisted of several steps. First, hydrogen permeability tests were carried out on both permeator units at temperatures up to 673 K, then breakthrough tests were performed on MeSMeR to define optimal operating conditions for the integrated configuration. Based on standalone results, a test matrix for the double-stage experiments was defined. Given the high permeation efficiency of the double-stage set-up, the measurement of the residual hydrogen content at the end of the process was obtained using a mass spectrometer opportunely calibrated and connected to the system. Across explored operating conditions, the double-stage system achieved hydrogen recovery efficiencies ranging from 99.9960% to 99.9998%. These outcomes provide a solid foundation for the sizing of hydrogen separation units responsible for treating the exhaust gas of plasma chamber in a fusion machine.

Design optimisation of a WCLL Test Blanket Unit for its testing in IFMIF-DONES

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The testing of Breeding Blanket (BB) mock-ups in an environment similar to that expected in future nuclear fusion reactors is a fundamental step towards the qualification of the technologies used in the different BB concepts and, finally, for the BB concepts themselves. Currently, the most advanced scientific program envisaging the testing of tritium blanket modules in a nuclear fusion environment is the ITER TBM (Test Blanket Module). Nevertheless, the reduced neutron fluence envisaged in new ITER baseline, reduces the scientific outputs that could be achieved with the TBM programme.

For all these reasons, it becomes important testing BB mock-ups also in other fusion-relevant machines, even though only partially representative of a nuclear fusion reactor. To this purpose, in 2025 ENEA and CIEMAT started an activity aimed at designing a Water-Cooled Lithium Lead Test Blanket Unit (WCLL-TBU) to be tested in IFMIF-DONES. Preliminary nuclear, thermal-hydraulic and thermo-mechanical analysis were carried out and a partial review of the design was performed.

This work reports on the design optimisation that has been carried after the preliminary design phase. In particular, the revised WCLL-TBU allows a maximum temperature lower than 550°C is achieved within both steel and breeder materials, ensuring a good representativeness of conditions expected in nuclear fusion reactors. Also the mechanical performances showed a beneficial effect of the new cooling system. Moreover, the most stressed areas have been checked against the design criteria of the RCC-MRx. Finally, the identification of interfaces with IFMIF-DONES Test Cell and auxiliary systems, as well as the WCLL-TBU requests in terms of coolant and purge gas thermal-hydraulic conditions has been carried out.

Thermal-Hydraulic Analysis of the EU-DEMO Primary Heat Transfer System with Different Steam Generator Options

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Efficient thermal power management represents a key requirement for the operation of the EU-DEMO fusion reactor, where the Primary Heat Transfer System (PHTS) acts as the interface between the Breeding Blanket and the Power Conversion System (PCS). Within the Water-Cooled Lead-Lithium (WCLL) concept, the PHTS is devoted for removing heat from the First Wall (FW) and Breeding Zone (BZ) using pressurized water, while ensuring thermal-hydraulic stability during the reactor characteristic pulsed operation.

The objective of the current work is to evaluate the thermal-hydraulic performance of the WCLL BB PHTS, comparing two Steam Generator (SG) technologies: the Once-Through Steam Generator (OTSG) and the U-Tube Steam Generator (UTSG). To achieve this goal, the Nuclear Engineering Research Group of Sapienza University of Rome, in collaboration with ENEA, has developed a comprehensive system-level model using the RELAP5/Mod3.3 code. The modeling approach integrates a detailed BB sector representation, including both Inboard and Outboard segments, into the complete PHTS circuit, which features eight semi-rings arranged into four primary coolant loops connected to a pressurizer. To optimize computational costs while preserving hydraulic accuracy, the 16 toroidal sectors were collapsed into a single representative component. A detailed nodalization was developed to accurately reproduce flow circulation and heat transfer phenomena within the system. To provide a realistic assessment of the plant lifecycle, the analyses were performed at both Beginning of Life (BoL) and End of Life (EoL), accounting for degradation effects such as tube plugging within the SGs.

The analysis includes steady-state verification at Full Plasma Power, operational pulse-dwell transients, and accidental scenarios, focusing on the Loss of Flow Accident (LOFA). For operational transients, different control strategies were investigated for the OTSG to maintain primary temperature stability by adjusting the secondary feedwater mass flow. In the accidental analysis, the system response was analyzed under different plasma ramp-down curves to identify critical safety margins.

Results demonstrate that both SG technologies are capable of meeting design requirements for heat removal under nominal conditions, with component aging having a limited impact on overall hydraulic behavior. Transient simulations highlight the importance of the BB thermal inertia, which delays heat transfer to the primary coolant. Safety analyses confirm that while both configurations can withstand a complete LOFA, the plasma power reduction rate strongly affects temperature and pressure peaks. Overall, this work supports the assessment of the current WCLL PHTS design, providing a technical basis for selecting the most suitable SG technology for future investigations.

W-HYDRA: An Integrated Large-Scale Experimental Platform for Thermal-Hydraulic and PbLi–Water Interaction Studies in WCLL Breeding Blanket Systems

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The Water-cooled lithium lead thermal-HYDRAulic (W-HYDRA) platform is a large-scale experimental infrastructure under construction at the ENEA Brasimone Research Centre to support the development and validation of the Water-Cooled Lithium Lead (WCLL) Breeding Blanket (BB) and its Steam Generator (SG) operation within the EUROfusion framework. The facility reproduces relevant thermal-hydraulic, pressure, and heat flux conditions for both ITER Test Blanket Module (TBM) and EU DEMO WCLL configurations.

W-HYDRA operates up to 2 x 1.7 MWe electrical heaters and 18.5 MPa design pressure and integrates three main experimental facilities: (i) the Water Loop (WL), (ii) the STEAM facility, and (iii) the LIFUS5/Mod4 loop, integrated by auxiliary systems for filling, drainage, pressure suppression, cooling, utilities, and a data acquisition and control system.

The WL is the core system: a pressurized primary water circuit delivering controlled mass flow (up to 22 kg/s at 15.5 MPa and 295°C) and thermal power to dedicated test sections. Primary, secondary and tertiary circuits (low power, low flow branch) ensure heat transfer and rejection. The loop is designed for rapid pulse–dwell–pulse transients representative of ITER operation and enables testing of scaled EU DEMO BB mock-ups and full-scale ITER TBM hydraulic layouts. To simulate plasma-deposited heat loads on First Wall and TBM mock-ups, the WL is coupled to a Vacuum Chamber equipped with an 800 kW Electron Beam gun. This configuration allows controlled high heat flux testing under vacuum conditions, enabling investigation of heat transfer coefficients, thermo-mechanical response, and cooling performance under reactor-relevant boundary conditions.

The STEAM facility supports experimental qualification of the pulsed operation of the WCLL SG, thanks a water/steam loop coupled to the high power, high flow branch of the WL and enabling steady-state and transient investigations, including pulse-dwell transitions.

The LIFUS5/Mod4 facility is a full-scale representation of the ITER WCLL PbLi loop. When integrated with the WL, it supports experiments on water–PbLi interaction phenomena, including in-box Loss Of Coolant Accident scenarios, allowing analysis of pressure transients, exothermic reactions and hydrogen production.

The integrated and modular architecture enables separate-effect and coupled experiments on water coolant systems, liquid metal loops, and high heat flux components within a single infrastructure, providing a platform for thermal-hydraulic model validation, component qualification, safety assessment, and technology advancement for ITER and DEMO.

This paper describes the plant architecture, executive design criteria, main engineering solutions, selected key components (pumps, valves, heat exchangers), and the construction programme and implementation phases.

Structural Simplification and Fabrication Procedure Development for the HCCP TBM Shield with Validation through a Small-Scale Mock-up

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The Helium Cooled Ceramic Pebble (HCCP) Test Blanket Module (TBM), jointly
developed by Korea and Europe, includes a highly integrated shield structure composed of multiple blocks and complex piping systems. The shield accommodates water cooling lines, helium inlet/outlet/bypass lines, purge gas lines, and feedthrough pipes for NAS instrumentation, resulting in significant structural complexity. The conventional design incorporates numerous stiffening plates, which limit welding accessibility, increase residual stresses, and complicate non destructive examination, thereby reducing manufacturability.

In this study, a structurally simplified internal configuration is proposed by reducing the number of stiffening plates while maintaining functional requirements. Structural analyses were performed to evaluate stress distributions under operational loading conditions, confirming that
the modified design satisfies structural integrity criteria with safety margins comparable to or exceeding those of the reference design. In parallel, an improved fabrication procedure was developed, including optimization of welding sequences and welding methods to enhance constructability and inspection accessibility.

To validate the proposed design and fabrication approach, a geometrically scaled small-scale mock-up is being designed and manufactured. The mock-up will provide experimental verification of manufacturability improvements and structural soundness. The present work contributes to improving the feasibility and fabrication reliability of the HCCP TBM shield for future fusion applications.

Development of a continuous, centrifuge-based, tritium-compatible Pellet Injection System

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

High speed injection of frozen deuterium-tritium fuel pellets is considered as one of the most promising plasma fueling technologies for fusion reactors based on magnetic confinement, most commonly Tokamaks and Stellarators. Numerous pellet injection systems have been designed and successfully tested in the past decades. Almost all these tests have been conducted without tritium, using pure deuterium or protium-deuterium mixtures. This is because the effort required for tritium tests is significantly higher due to the cost of tritium, tritium handling and containment, and material compatibility. However, the tritium beta decay is expected to significantly alter its behavior at cryogenic temperatures when compared to protium and deuterium. All in all, this has resulted in a situation where the principle is well understood for inactive operation but not yet designed and qualified for the use with tritium. The roadmaps of several private and public institutions that aim to build and operate fusion pilot plants in the 2030s now require that the technology readiness level (TRL) is matured.

Kyoto Fusioneering Europe GmbH (KFEU) and the Karlsruhe Institute of Technology (KIT) are jointly developing a demonstration unit for a continuous centrifuge-driven pellet injection system (PIS). The system is designed to be fully tritium-compatible and will undergo tritium testing at UNITY-2. This contribution will introduce an overview of the key characteristics of the PIS, the current state of development and the planned test program.

Design Improvements to the HUN-REN CER Hydrogen Batch Extruder and Systematic Measurements for Extrusion Model Validation

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Pellet injection technique is a proven, leading technology for fuelling and controlling the plasma density in magnetically confined fusion devices. To minimise plasma perturbations upon injection, fuelling pellets are small, typically a few mm in diameter and length, necessitating moderate injection frequencies (10–20 Hz) and quasi-continuous operation.
Fuelling pellet injection systems consist of three main components: an ice extruder, a pellet cutter, and an accelerator. At the outlet of the extruder, a solid hydrogen rod is produced, which is cut to predefined lengths by a puncher or cutter mechanism. The pellets are then launched by gas propulsion or a centrifuge.
In the Fusion Plasma Physics Department of HUN-REN Centre for Energy Research, a new pellet puncher for EU DEMO was developed, in collaboration with KIT. Additionally, a batch-type cryogenic extruder that would be used to test the pellet puncher was developed by HUN-REN CER. The batch extruder has a nominal ice capacity of >5,000 mm3 and is designed for an extrusion duration of 10 seconds at 15 Hz injection frequency. The solid hydrogen rod produced by the extruder is 3.2 mm in diameter, and the target flow rate is 386 mm3/s, which is required for the 15 Hz operation of the pellet puncher when the pellet length is 3.2 mm.
The extruder is installed in a custom-made, large flexible cryostat and is cooled by a two-stage cryocooler. The second stage of the cryocooler is connected to the extruder, while the first stage is used for feed gas pre-cooling and cooling the support structure. The heat shield covering the extruder is cooled by liquid nitrogen to reduce the thermal load on the cryocooler. During the extrusion process, temperatures at multiple points, heating power, extrusion force, and piston position are all measured.
At the conference, the design upgrades to the batch extruder will be presented, along with thermal simulations of their effects and systematic measurement results that serve as a basis for extrusion model validation.

Keywords: Solid Hydrogen, Extrusion, Cryogenic Extruder, Plasma Fuelling

Accumulation of Tritium and Protium in the Reactor Fuel Cycle Operation with Direct Internal Recycling

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

In this article, we estimate the inventories of tritium and protium in the fuel cycle of a fusion power reactor operating with direct internal recycling (DIR), where the exhaust gas mixture is returned to the reactor chamber after purification. It is shown that in operation with DIR the burning fraction increasing with increasing the separation fraction and concequently decreasing the tritium inventory.

In expected D-T imbalance in the reactor plasma due to concomitant fusion reactions the maximum burning fraction can be achieved with a lower tritium concentration than the deuterium concentration. This requires a lower concentration of tritium in the fuel supplied to the reactor from an external source. It is shown that the allowable protium concentration in the main plasma of the DEMO with a helium concentration of 5-7%, and xenon concentration of about 0.036% must not exceed 7%.

Cryogenic pellet injector development for large fusion devices

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

A cryogenic pellet injector system for fuelling pellets, made of Protium or Deuterium, was developed at the Fusion Plasma Physics Department (FPL) of HUN-REN Centre for Energy Research. The pellet injector consists of a mechanical extruder, a pellet cutter and an accelerator system – all of which was developed in-house.

The pellet extruder is a piston-type batch extrusion system, with a capacity of ca. 5300 mm3 Hydrogen ice, equivalent to an extruded rod length of ca. 50 cm for DEMO-relevant pellets (rod diameter of 3.2 mm), and 1 m for JT-60SA-relevant pellets (rod diameter of 2.4 mm). The design is based on a previous device developed for DIPAK [1]. The batch extruder can also be equipped with an additional storage and feeder unit, where the ice rod is advanced purely mechanically, allowing for precise timing down to the ms range during pellet injection.

Several pellet cutter designs were investigated: for the batch extruder with a storage unit, where the ice rod advancement is also step-wise (just as the cutting), a single-action, electromechanical cutter-puncher design was used. In this case, the rod advancement and the pellet cutting are consecutive actions, hence the movement of the ice rod is not hampered by the cutter blade. In case of a continuous pellet source, such as a screw-type extruder, the movement of the ice rod is uninterrupted; hence, a similar cutting mechanism where the cutter blade can obscure the ice rod movement is not the optimal solution. Therefore, a double-action cutter mechanism was designed, where the cutter blade blocks the ice rod for an order of magnitude less time than in the case of the single-action cutter. On the other hand, pellets are punched in two directions, therefore a collection track is also needed to guide all pellets to the same exit. The effect of the collection track on the jitter of the pellet arrival time – critical for the centrifuge-based accelerator – was also investigated.

The pellet accelerator system is based on a small centrifuge “TATOP” [2], previously used for solid state pellets. The TATOP features a main acceleration arm with a radius of only 10 cm, allowing for a maximum pellet speed of 450 m/s, while still using standard EN AW 6082 aluminum as base material.

[1] B. Longauer et al, ISFNT-16, Knoxville, USA, 2025
[2] T. Szepesi et al, Fusion Engineering and Design 96–97 (2015) 707–711

Integration of On-The-Fly Variance Reduction Method into OpenMC and Application to EU-DEMO Fusion Reactor

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

As a rapidly advancing open-source Monte Carlo code, OpenMC has shown great potential in fusion neutronics. However, its variance reduction techniques for fusion device shielding calculations remain limited, which restricts its efficiency in addressing deep-penetration shielding problems common in fusion reactor analysis.

The breeding blanket is a critical functional component of fusion power plants, responsible for tritium breeding, energy extraction, and radiation shielding. Comprehensive neutronics analysis is essential for the design and safety of fusion reactors. Beyond the core neutronics evaluation of the blanket itself, shielding calculations for irradiation-sensitive components are equally crucial to ensure reliable operation. These shielding calculations rely heavily on advanced variance reduction techniques, among which the MCNP-based On-The-Fly (OTF) global variance reduction method has been widely applied to major fusion devices such as ITER and CFETR.

This work develops and implements the OTF method in OpenMC. The modified OpenMC code is then applied to the neutronics analysis, including shielding calculations, for the EU-DEMO fusion reactor. Simulation results are systematically compared with those obtained from MCNP to further benchmark and verify the efficiency and accuracy of OpenMC-OTF.

Conceptual Cryopump for Removing Impurities and Helium-Ash for a Fusion Fuel Cycle

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

When it comes to exhaust puming in fusion reactors, cryopumps are the de facto solution. This is because they offer high pumping speeds at relatively low cost owing to their simplicity, compared with mechanical vacuum pumps. In addition, cryopumps can remove/separate unwanted gas constituents from the fusion exhaust while actively cryopumping. A continuously regenerating cryopump provides uninterrupted pumping and gas separation. For example, the snail pump has been demonstrated as a continuously regenerating cryopump, delivering a significant throughput, ~50 Pa.m^3/s, and enabling helium separation. The snail pump maintains cryo-surfaces at temperatures below the saturation temperature of the DT fuel and above that of helium, thus capturing DT fuel while allowing the helium to pass through. A snail pump equipped with an impurity-removal cryopump (for impurities other than helium ash) should provide complete fusion exhaust processing, except for neon separation if neon is used as a divertor seeding gas.

In this work, we present a conceptual pump consisting of a batched impurity-removal cryopump in tandem with a continuously regenerating snail pump as a complete fusion exhaust processing system. The impurity pump section is designed to remove all impurities from DT stream except helium ash, which is separated by the snail pump. The snail pump section provides primary pumping and is optimized for cryopumping performance and fuel capture. Here, we present a summary of the computational studies performed for this conceptual pump.

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Experimental Evaluation of a Reactive Molecular Sieve Bed Mock-Up for Tritium Recovery via Isotopic Exchange

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The effective recovery of hydrogen isotopes is a fundamental requirement for purge gas processing systems in fusion breeder blanket technologies. This study reports experimental results on the isotopic exchange process carried out in a reactive zeolite bed mock-up using deuterated water and hydrogen as working media. Tests were conducted on a 1:10 scale Reactive Molecular Sieve Bed (RMSB) packed with platinized zeolite containing 0.3, 0.5, and 0.7 wt% platinum. Following the adsorption of approximately 600 g of deuterated water, isotopic exchange was initiated by passing hydrogen through the heated bed at temperatures between 25 °C and 120 °C. The deuterium concentration in the outlet gas stream was continuously measured by quadrupole mass spectrometry, and extraction efficiencies were evaluated over a standard exchange duration of 10 hours. The findings indicate that both elevated temperatures and higher platinum loadings enhance exchange performance significantly. Above 80 °C, the system exhibited a quasi-steady exchange regime, suggesting diminished kinetic and mass-transfer constraints. Considering the balance between performance and material cost, a platinum loading of approximately 0.5 wt% emerges as a practical and efficient option for tritium recovery system implementation.

CFD-based thermal analysis and heat transfer enhancement of a ZrCo packed getter bed

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Getter beds based on zirconium–cobalt (ZrCo) hydrides represent a promising alternative to cryogenic molecular sieve beds for hydrogen isotope removal in helium purge loops of fusion blanket tritium extraction systems. However, regeneration of ZrCo getter beds requires heating to approximately 400 °C under vacuum, where convective heat transfer becomes negligible and thermal performance is limited by the inherently low effective conductivity of packed hydride material. In this work, a detailed CFD-based thermal assessment of a modelled ZrCoHx packed getter bed was performed using ANSYS Fluent, focusing on transient heating behavior, temperature gradients, and effective heat transfer coefficients during regeneration by hot helium gas circulation through an internal heating pipe. The packed bed was modelled as a homogenized solid medium using experimentally derived effective thermal properties for ZrCoHx chunk/flake material. Several design enhancement strategies were investigated, including the addition of 10% and 20% vol. copper spheres, the integration of sintered stainless-steel fins, and the use of high-conductivity copper fins with a copper support pipe. Simulations were carried out for helium inlet temperatures of 450 °C at flow velocities of 2 m/s and 10 m/s, assuming adiabatic external boundaries and negligible heat losses. Results indicate that the baseline ZrCoHx bed exhibits strong radial temperature gradients and slow thermal equilibration due to limited heat propagation through the porous packed structure. Thermal enhancements significantly improve heating performance: copper sphere additions increase the steady-state effective heat transfer coefficient, while finned copper structures provide the fastest transient heating and improved temperature uniformity by establishing efficient conductive pathways across the bed. Increasing helium velocity improves initial heating rates but has a secondary effect compared to structural conductivity enhancements. The study demonstrates that optimized internal thermal design is essential for reducing regeneration time and ensuring uniform temperature distribution in ZrCo getter beds intended for fusion tritium extraction applications.

Thermomechanical Design Activity for the Development of the Alternative Water-Cooled WLCB Breeding Blanket concept for DEMO

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

In view of the definition of the driver blanket for the EU-DEMO fusion reactor, three Breeding Blanket (BB) concepts have been identified as baseline BB concepts for a DEMO reactor: the Water Cooled Lead-Lithium (WCLL), the Helium Cooled Pebble Bed (HCPB) and the more novel Water cooled Lead and Ceramic Breeder (WLCB).
The WLCB is a relatively novel addition to the EU BB baseline. This concept mixes the strong points of WCLL and HCPB concepts, namely the use of water as coolant medium and the use of ceramic pebbles as tritium breeder and an in-situ tritium extraction with a purge gas within the BB. Moreover, to overcome the technical, safety and economic issues associated with beryllium based neutron multipliers, the adoption of (molten) lead in canisters is foreseen.
This emerging concept has evolved through several designs in order to improve its performances from both the neutronic and thermomechanical points of view. One of the key points is the mechanical strength under breeding zone over-pressurisation operating conditions, as a consequence of an in-box loss of coolant, which has driven this new design.
This paper deals with the thermomechanical assessment of the 2024 design using RCC-MRx rules. A specific two-step iterative method has been set up, which allows the main key cooling parameters to be defined first on a sub-model and then applied to the full blanket geometry. It allows to improve the iterative process to define the final geometry. Finally, some design improvements have been proposed.

#13160

A preliminary study of blanket systems for laser fusion reactors toward achieving a high tritium breeding ratio

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Fusion research is transitioning toward the reactor development phase; however, blanket systems remain at an early stage of development. Several blanket concepts have been investigated for magnetic confinement fusion reactors. In such systems, the blanket is installed within the limited space between the superconducting magnets and the fusion plasma to breed tritium. Achieving a tritium breeding ratio (TBR) greater than 1 is a fundamental design requirement. Magnetic fusion blankets typically employ lithium enriched in 6Li to attain high breeding performance. However, the use of 6Li-enriched lithium substantially increases reactor operating costs due to its high cost. In contrast, blanket systems for laser fusion reactors have not yet been fully designed and require design approaches distinct from those of magnetic fusion systems. The larger blanket volume available in laser fusion reactors provides greater design flexibility. Consequently, a high TBR may be achievable using lithium with its natural isotopic composition. This study presents a preliminary investigation of blanket systems for laser fusion reactors, with emphasis on tritium breeding performance. Simulation results demonstrate that a TBR exceeding 1.5 can be achieved in laser fusion blanket systems.

Thermal–Hydraulic Assessment of an Alternative Manifold Design Approach for the HCPB Breeding Blanket

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The Breeding Blanket (BB) represents one of the most critical components foreseen for the European DEMO reactor, currently under development within the European fusion roadmap. Its primary functions include ensuring tritium self-sufficiency, extracting nuclear heat for power conversion, and providing shielding to protect the vacuum vessel and superconducting magnets. Among the candidate blanket concepts under investigation, the Helium-Cooled Pebble Bed (HCPB) design employs high-pressure helium as coolant and ceramic pebble beds as tritium breeders, combined with neutron multiplier materials. The new HCPB BB design consists of a pin-based architecture. Each pin integrates helium channels for cooling, lithium-based Advanced Ceramic Breeder acting as tritium breeding material together with lead neutron multiplier elements, which enhance neutron utilization and sustain an adequate tritium breeding ratio. This heterogeneous internal structure leads to complex thermal fields and cooling channel geometries. The thermo-hydraulic performance of the coolant distribution system, including the manifold region, plays a strategic role in this framework.
This work presents a thermal–hydraulic assessment of a revised manifold section developed to improve flow uniformity among the different pin units, and to simplify the geometry, reducing the number of welds required. A computational fluid dynamics (CFD) analysis of the manifold and connected cooling network was carried out to assess the helium flow distribution and the overall pressure losses of the manifold layout. The simulations consider high-Reynolds-number turbulent helium flow, representative inlet mass flow rates and outlet pressures, with dedicated mesh refinement strategies to ensure accurate boundary layer resolution.
In parallel with the CFD activities, a system-level model of the manifold section was developed using the thermal-hydraulic system code RELAP5 Mod.3.3 to provide complementary one-dimensional analyses. The RELAP5 model was implemented to reproduce the relevant hydraulic paths and pressure losses of the manifold and connected cooling channels. The system-code simulations supported the design phase by enabling rapid parametric assessments and by providing an independent benchmark for the CFD results. The comparison between detailed CFD predictions and RELAP5 outcomes allowed the qualification and calibration of the one-dimensional model, strengthening confidence in its use for design support and future transient investigations. The combined use of high-fidelity CFD analyses and system-level modelling provides a consistent and scalable methodology for manifold optimization within the HCPB BB framework.

Performance assessment of a Quick Disconnecting System sealing junction for the ITER Test Blanket Module Helium Cooling System

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

One of the key challenges in the TBM program is ensuring the integrity and sealing performance of the Helium Cooling System (HCS), which is essential for maintaining operational efficiency and safety. The HCS must withstand extreme conditions, including high temperatures up to 500°C and pressures up to 9.8 MPa. In order to perform the replacement operation of the Port Cell equipment, those lines will need to be dis-connected and re-reconnected several times. Therefore, the possibility of using metallic sealing junction is being investigated. Hence, robust validation of the sealing performance is required to guarantee adequate reliability. On top of that, the TBM must be designed to ensure that occupational radiation exposure is kept As Low As Reasonably Achievable (ALARA), especially for workers in port cells.

To address these challenges, a bespoke sealing solution using a Helicoflex® metallic seal with a 50µm silver coating compressed between two flanges by a Quick Disconnecting System (QDS) collar has been proposed. This junction type aims at providing a reliable and efficient sealing mechanism capable of withstanding the harsh operational conditions of the HCS while improving the intervention time compared to traditional bolted flanges.

A mock-up was designed and manufactured to explore the feasibility of using those sealing junctions for the DN80 HCS in representative conditions. The mock-up includes similar dimensions and materials, with a focus on validating the sealing performance under maximum temperature and pressure targeted by the ITER TBM program. With the same ambition of representativeness, the mock-up was equipped with a heating device so that heat is supplied from the inside of the mock up.

With this experimental setup, comprehensive testing and validation of the sealing solution under high temperature and pressure conditions were performed. Five seals were tested, including three under nominal maximum conditions of 500°C and 98 bars. For every seal tested, a leak rate of less than 3.0×10^(-9) Pa.m^3.s^(-1) was measured, which meets both the tritium contamination criteria and the VQC3 standard.

“The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.”

A Multi-physics Modelling and Simulation Platform for Breeder Blanket Technology

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Within fusion power-plants, the breeder blanket in particular is an inherently coupled system, having multiple competing functional requirements. The ability to efficiently extract heat and tritium must balance the demand for component lifetimes that satisfy a cost-effective operational schedule of availability and maintenance. Modelling these phenomena requires simultaneous treatments of particle transport, transmutation, speciation, thermodynamics, solid mechanics, fluid dynamics, and material behaviours. This implies an exchange of data between domain-specific physics models over appropriate length and time scales to attain a self-consistent solution. Moreover, the evolution of coupled variables may give rise to unexpected emergent behaviour that cannot be captured without an integrated model.

In an effort to derisk tritium breeding technology, the UKAEA's Lithium Breeding Tritium Innovation (LIBRTI) program is delivering a facility that will host a 14 MeV neutron source and install several modular experiments. Furthermore, to enable virtual representations of the breeding experiments, a flexible and scalable multi-physics modelling and simulation platform, is currently under development. This activity will facilitate a deeper understanding of the mechanisms that underpin observations of tritium production and extraction, provide a quantitative assessment on the validity of the models, and a route to accelerating innovation with in-silico design.

The Multi-physics Object-Oriented Simulation Environment (MOOSE) software has been selected to provide the base functionality. To supplement its existing physics domains, new interfaces to FISPACT-II and FLUKA, which enable modelling of inventory evolution and charged particle transport respectively, have been developed. To demonstrate current capabilities, three cases spanning the potential breadth of experimental scenarios have been modelled. The first case is a conceptual solid breeder experiment, consisting of a tessellated assembly of pins aligned with the neutron source. For the second case of molten salt breeders, the LIBRA and BABY-LIBRA experiments have been considered. Finally, for the liquid metal case, we have modelled the TRIEX-II Gas-Liquid Contactor. In all cases a multi-physics analysis has been performed, and the implications for tritium extraction are explored. We also explore strategic opportunities to validate these multi-physics models in the lead-up to the delivery of the LIBRTI facility. This approach should enable maximisation of the information to be gained from subsequent experiments and provide confidence in the use of this multi-physics software for the design of breeder blankets.

Thermo?Mechanical Analysis for HCCP Ancillary Systems (HCS & CPS) in the TCWS Vault Area

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

This paper presents the methodology and analysis plan for thermo-mechanical (TM) calculations in support of the helium-cooled ceramic pebble (HCCP) test blanket system. The focus is on the helium cooling system (HCS) and coolant purification system (CPS), which are installed in the tokamak cooling water system (TCWS) vault (11_L4_04). The objective is to reduce the risk of integration by informing strategies for the placement, routing and support of equipment, and by establishing a traceable basis for verification ahead of design. The scope covers TCWS Vault Area layouts and interfaces with CCWS/CHWS, as well as port cell connection piping and the configuration of large thermal masses, which can lead to differential expansion and stiffness discontinuities that are relevant for managing support and nozzle loads.

The analysis framework uses vault-scale piping and secondary steel models implemented in ROHR2. These models are progressively matured through geometry import, secondary structure definition, and iterative primary support definition. This enables early screening while boundary conditions, masses, and allowables are refined. Code screening is applied using ASME B31.3 (2024), structuring verification into (i) internal pressure checks and (ii) load category checks, which are consistent with sustained, thermal and occasional groupings. The load cases follow the project requirements and prioritise the Normal Operating State (NOS), the Tritium Outgassing State (TOS) and seismic demand (including higher-level screening cases, such as SL-2), in order to drive early iteration of the layout and support. For the preliminary definition of the envelope, the design values for piping consider 101 bar and 550 °C. There is an explicit plan to iterate the design pressure/temperature to reflect the maximum operational NOS/TOS envelope, where appropriate, to achieve design convergence.

The results already show the need to converge design envelopes and restraint philosophies through targeted iterations of routing, supports and, where required, pipe schedules. Planned outputs include a traceable verification package, demand/utilization maps to prioritize hotspots, support and spool detailing rules, and interface reaction envelopes to guide equipment and TCWS platform integration.

Keywords: Fusion technology; Tritium fuel cycle; pipe-stress analysis.

Integrated Assembly and Maintenance Planning for the HCCP HCS and HCCP CPS Ancillary Systems in ITER Based on RAMI Driven Design

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

This paper reports the approach and outcomes of an integrated planning study for two ancillary systems of the HCCP breeding blanket for ITER—the Helium Cooling System (HCS) and the Coolant Purification System (CPS). The work is motivated by the need to de risk activities on a fusion installation and to embed maintainability-by-design while preserving availability for early operation. The works couple the assembly plan with the maintenance plan and trace both to a previously completed RAMI analysis focused on critical components and availability drivers.

The method uses model-based definition and interface mapping to derive assembly sequencing from functional boundaries, cleanliness constraints, isolation logic, and access envelopes. In parallel, the maintenance plan is structured around RAMI-identified critical items and plausible failure/inspection vectors (e.g., helium circulators or safety isolation valves), producing detailed procedures, required tools and fixtures, and commissioning/verification hooks that are consistent with the installation sequence. The paper details how assembly choices—such as routing and modularization, connector and joint strategies, placement of isolation and sampling points, and provision for online diagnostics—directly shape maintainability, turnaround times, and spare-part strategies.

Representative installation sequences for the HCCP HCS and HCCP CPS subassemblies—together with pre commissioning activities aligned with component qualification and confinement requirements—are detailed, providing a complete view of how these ancillary systems are brought to operational readiness. RCM approach explicitly, based on RAMI results, linked to the selected maintenance concepts (preventive, predictive, and corrective), forming the basis for design feedback loops that refine system architecture and operational procedures. This integrated approach yields installation and maintenance strategies that minimize worker dose and room occupancy, while defining spares and tooling philosophies focused on the dominant availability drivers.

The work delivers finalized assembly and maintenance plans for both HCCP-HCS and HCCP-CPS, a traceability matrix connecting RAMI findings with planning decisions, and consolidated lessons learned on balancing integration simplicity with maintainability in breeding blanket ancillary systems—thereby providing a practical and transferable template for maturing ancillary system designs toward higher availability.

Keywords: Fusion technology; Tritium fuel cycle; assembly planning; maintenance planning; RAMI; design-for-maintenance.

Design and Development of the TRYDENT Facility for Trace Hydrogen Purification

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The fuel cycle architecture of a fusion machine must be able to process tritiated gas streams under highly variable operating conditions spanning wide ranges of pressure, temperature, flow rate, and tritium concentration. To accommodate this diversity, multiple purification technologies are employed, including permeators, wet scrubber columns, copper-oxide/molecular-sieve (CuO-ZMS) systems, and getter materials. Permeators are the reference solution for treating hydrogen-rich gas streams, while the remaining technologies are effective for removing hydrogen isotopologues present at trace levels. Wet scrubber columns, in particular, are widely used as final tritium barriers due to their continuous operation, robustness, and reliability, having as drawback the generation of large quantities of low-activity tritiated water. The CuO-ZMS process also produces tritiated water, although in substantially smaller volumes and with higher tritium concentration compared with wet scrubbers. Getter beds represent another promising option, however, their current use is largely confined to vacuum applications.

Focusing on the application of CuO-ZMS, and getter beds, this work presents the design and procurement of a dedicated experimental apparatus named TRYDENT (TRace hYDRogEN purification). TRYDENT comprises three heated beds, each capable of reaching temperatures up to 1100 K and built with identical architecture. Key features include three independently controlled heating zones, a pre-heating section, a central reaction chamber, bottom gas inlet and top outlet for both adsorption and regeneration operations, a stainless-steel sintered diffuser to ensure uniform gas distribution, and a mechanical grid to prevent material displacement. The design is supported by detailed calculations involving pressure drop prediction, spatial velocity, void fraction, mass transfer zone estimation, pellet inventory, regeneration time, and overall bed sizing.

TRYDENT is a custom built, high performance Sieverts-type facility developed at ENEA Frascati to bridge the technological gap identified through a market survey, which revealed that commercial systems are constrained by limited maximum operating temperatures and insufficient sample holder volumes. The facility will enable comprehensive characterization of materials relevant for trace hydrogen removal across the broad operating conditions required in fusion fuel cycle applications. Furthermore, TRYDENT supports multiple operating modes, allows precise studies of regeneration procedures, and offers the possibility to investigate integrated solutions.

#13210

First Insights from the “Advanced Ceramic Breeder's in situ Tritium Release Experiment under Neutron irraDiation” (ACB-TREND) campaign

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Advanced Ceramic Breeder (ACB) pebbles, consisting of the two phases Li_4SiO_4 and Li_2TiO_3, represent the European reference material for solid breeder blanket concepts for the ITER Test Blanket Module and DEMO. While its material properties have been extensively investigated out of pile, experimental data under neutron irradiation remain very limited. To specifically address the ACB’s tritium release in situ performance, the Advanced Ceramic Breeder’s in situ Tritium Release Experiment under Neutron irraDiation (ACB TREND) campaign was initiated.

Experiments were conducted at the WWR K research reactor in Almaty, Kazakhstan, using a dedicated irradiation rig designed to provide fusion relevant conditions. With that, a tailored neutron spectrum is achieved by using cadmium shielding to suppress thermal neutrons and to limit lithium burn up. The irradiation campaign was performed in two runs with lower and higher irradiation temperatures. The tritium release was measured in situ using two ionisation chambers to detect the total amount of tritium and the dry tritium species, respectively. During the irradiation experiments, the irradiation temperature (400–900 °C) and the composition of the purge gas (He + x vol% H2, where x = 0, 0.01, 0.1, and 0.2) were varied.

This contribution reveals initial insights from the ACB TREND campaign, focusing on the in situ tritium release behaviour of ACB pebbles influenced by different irradiation temperatures and purge gas compositions. Preliminary results on the tritium residence time, determined from definite temperature levels, will be shown. The presented results will contribute to the qualification of ACB pebbles for their use in solid breeder blankets.

Fluid dynamics simulation of phenomena related to molten salt use in nuclear fusion

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

This work presents a computational investigation of the thermo-fluid dynamic behavior of molten-salt cooling systems for next-generation nuclear fusion reactors, with emphasis on the ARC (Affordable Robust Compact) concept. The study examines heat-removal strategies in two key components—the Vacuum Vessel (VV) and the divertor—through the application of Passive Heat Transfer Enhancement Techniques (PHETs) in rectangular cooling channels operating under fusion-relevant conditions. The coolant considered is FLiBe, a high-Prandtl molten salt used simultaneously for heat extraction, breeding, and shielding, whose flow behavior is strongly affected by magnetohydrodynamic (MHD) effects.

Three-dimensional CFD simulations were performed using ANSYS Fluent, incorporating both turbulence modeling (RANS with k-w and RSM formulations) and magnetic-field-induced MHD phenomena. Among the PHETs evaluated, V-shaped ribs (V-ribs) emerged as the most effective geometry, outperforming twisted tapes and wire coils due to their ability to generate structured secondary vortices and intensify near-wall turbulence. Model validation confirmed that RSM approaches better capture anisotropic turbulence and MHD-driven laminarization.

Two cooling-loop configurations were examined: (i) separated channels for VV and divertor, with shared total mass flow, and (ii) a single-channel layout connecting both components. In the first configuration, optimal V-rib geometries—specifically 90° ribs with a 10-mm pitch—yielded substantial improvements in heat transfer and reduced peak temperatures in the VV, maintaining Inconel-based structures within allowable limits. Parametric variations of mass flow indicated that moderate reductions in VV flow could benefit divertor cooling while preserving acceptable VV temperatures.

Conversely, divertor thermal management remained significantly more challenging. Even with optimized ribbed channels and increased mass flow rates, tungsten temperatures at the first wall exceeded design constraints, indicating that the divertor requires considerably higher average heat-transfer coefficients than those achievable with the tested PHETs. This highlights the need for advanced geometries, increased coolant flow, or operational adjustments to plasma exhaust conditions. Variations in the heat-transfer coefficient between FLiBe pool and Inconel showed limited effects, primarily in VV regions.

A preliminary MHD analysis revealed substantial increases in pressure drop in both plain and ribbed channels, while heat-transfer degradation was unexpectedly limited for ribbed configurations, suggesting that PHET-induced turbulence mitigates MHD laminarization effects.

The results demonstrate that optimized V-rib configurations can meet VV cooling requirements, whereas divertor cooling remains a critical bottleneck, necessitating further innovation in PHET design, coolant routing, or plasma-surface interaction management. These findings contribute to integrated thermohydraulic optimization strategies necessary for reconciling material constraints with compact fusion reactor design requirements.

#13225

Tritium Production from Helium-3 Using Inertial Electrostatic Fusion Devices

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Helium-3 is the product of tritium decay. With a future fusion industry dependent on fine tritium breeding margins with significant inventories, it is important to assess means to minimise losses from an inventory. The relatively short half-life of tritium causes losses of ~5% per year, which must be accounted for by additional breeding. While it is possible to achieve this with further throughput of lithium in breeder blankets, the ‘free’ helium-3 produced in this decay can be readily converted back to tritium with thermal neutron radiation, which can be done separately from a full breeder blanket.
In this work, an inertial electrostatic confinement (IEC) fusion neutron source capable of deuterium-deuterium and deuterium-tritium operations, was used to produce tritium from helium-3 gas under various conditions. The resulting tritium in the gas was analysed semi-quantitatively by on-line residual gas analysis quadrupole mass spectrometry, and quantitatively by off-line liquid scintillation counting.

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#13253

Structural Evaluation of the HCCP TBM-Set under Thermo-Mechanical Load

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

This study presents an ongoing structural assessment of the ITER Helium-Cooled Ceramic Pebble Bed (HCCP) Test Blanket Module (TBM) set using ANSYS-based finite element analysis in accordance with RCC-MRx design criteria. The objective of the study is to evaluate the structural integrity of TBM components under representative operating and accidental conditions relevant to ITER application. Thermo-mechanical load combinations have been established by considering internal pressure, thermal gradients induced by plasma operation and baking scenarios. For selected operating conditions, detailed stress distribution and deformation behavior of the TBM structural components, including the shield and attachment structures, were investigated. The evaluation was performed based on RCC-MRx rules for stress categorization and allowable limits. Preliminary results indicate that, for the analyzed load cases, the maximum equivalent stresses remain within the allowable design limits defined by RCC-MRx. Localized stress concentrations were identified near structural discontinuities and support regions. Additional evaluations, including electromagnetic transient loading scenarios and comprehensive load case combinations, are currently in progress to complete the overall structural qualification process. The study establishes a systematic analysis framework for RCC-MRx-based structural integrity of the HCCP TBM set and provides a technical basis for ongoing design optimization.

Hydrogen Isotope Recovery from Supercritical Carbon dioxide Cycle in Fusion Reactor

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

This study considers the use of supercritical carbon dioxide (sCO_2) as a heat transport medium in deuterium–tritium (DT) nuclear fusion reactors, with a focus on detritiation in the cooling loop. Tritium, a key fusion fuel, may permeate through structural materials into the sCO_2 system. To capture it, an effective method is to oxidize tritium into a water isotope and subsequently adsorb it using a molecular sieve (zeolite MS-3A). Supercritical carbon dioxide is a promising coolant for blanket systems, both solid and liquid breeders, due to its low magnetohydrodynamic (MHD) effect and the expected ease of tritium contamination management. In addition, previous studies have shown that molecular sieves are a favorable option for the capture and immobilization of tritiated water; however, their application under supercritical CO_2 conditions remains insufficiently explored. Therefore, sCO_2 is assumed to be the coolant in this research, and a MS-3A is examined as a recovery method for tritium permeating into the coolant cycle.

The experimental setup, demonstrated in a non-radioactive environment, is based on a natural thermal convection loop operating at a maximum pressure of 15 MPa, with a high-temperature section at 500 °C and a low temperature section incorporating a zeolite-packed bypass line. The content of hydrogen compounds in sCO_2 is analyzed over time using quadrupole mass spectrometry (QMS). QMS measurements at m/z = 18 were performed to evaluate water related signals in the molecular sieve bypass line. A controlled amount of water was introduced into the system to establish a baseline H_2O signal, after which the gas stream was passed through a MS-3A bed under sCO2 conditions.

The QMS measurements showed a clear reduction in the H_2O signal by approximately 75% compared to the baseline, indicating effective water adsorption by MS-3A. Subsequently, desorption was carried out by heating the molecular sieve under a helium purge, resulting in an increase in the H_2O signal exceeding the baseline level. These results demonstrate reversible water adsorption–desorption behavior and validate the feasibility of using molecular sieves for hydrogen isotope management under sCO_2 conditions.

Mechanical strength and chemical stability of Li_{2+x}(Ti_zZr_{1?z})O_{3+y} breeder pebbles at elevated temperatures in 0.1% H_2/He

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

In thermonuclear fusion reactors that utilize the tritium-deuterium fusion reaction, tritium, which is rarely present in nature, must be produced by the nuclear reaction of neutrons with lithium-6 (^6Li) in the blanket surrounding the plasma. Lithium titanate-based ceramics, such as Li_2TiO_3 and lithium-doped Li_{2+x}TiO_{3+y}, have been extensively studied as potential tritium breeding materials. However, they suffer from microstructural changes and compositional changes due to lithium vaporization at high temperatures. To address this issue, Li_{2+x}(Ti_zZr_{1-z})O_{3+y} (LTZO), a solid solution of Li_{2+x}TiO_{3+y} and Li_2ZrO_3, was developed to improve high-temperature stability. The objective of this study is to clarify the lithium vaporization behavior and mechanical strength properties of LTZO at high temperatures in 0.1% H_2/He, simulating the actual operating conditions of tritium breeding materials. Annealing tests of LTZO were conducted in the temperature range of 320 °C to 900 °C for up to 2000 hours. The lithium vaporization rate from LTZO was confirmed to be lower than that from Li_{2+x}TiO_{3+y}, indicating improved high-temperature compositional stability. The crystal structure remained stable up to 600 °C, whereas phase transition and phase separation were observed at temperatures above 700 °C. Furthermore, mechanical strength increased at temperatures exceeding 800 °C. SEM observations revealed grain refinement above 800 °C, which is considered to contribute to the observed enhancement in mechanical strength.

Development of an Online Hydrogen Isotopes Multipoint Measurement System Using Raman Spectroscopy

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Real-time measurement and analysis of hydrogen isotope gas composition are essential for operational control of the fuel cycle and tritium inventory management in fusion plants. Kyoto Fusioneering (KF) is collaborating with the National Institute for Quantum Science and Technology (QST) to develop a measurement system capable of detecting hydrogen isotopes including tritium with minimal temporal gaps and a fast response time of less than one minute.

KF and QST have developed a Raman analysis system equipped with a semiconductor laser, a peltier-cooled CCD detector, and a newly developed optical system. This newly developed Raman analysis system, utilizing an optimized optical system, enables the online measurement and analysis of hydrogen isotope gas partial pressures down to several tens of Pa within one minute, with virtually no measurement gaps. It can measure and analyze up to four measurement points simultaneously per system. The Raman optical cell, designed for installation in process gases, was developed for use in high-concentration tritium processes. It uses no organic materials in gas-contacting parts, and all seals are metal. Tritium durability tests were conducted on the gas-contacting materials used in the optical cell, confirming its suitability for use in tritium environments. This optical cell is operable within a pressure range from vacuum to several hundred kPa. The use of a multiple reflection mechanism within the optical cell enables a several-fold increase in irradiation intensity, allowing the use of a semiconductor laser with sufficient output that does not require water cooling. An optical design that maximizes the collection efficiency of Raman scattered light, combined with the use of a special collimator, suppresses stray light generated by laser irradiation within the cell. This allows for highly efficient collection of Raman scattered light without requiring special filters in the wavelength range necessary for measurement, keeping the impact of Rayleigh scattered light on the baseline extremely low. Preliminary tests using tritium confirmed the sensitivity configuration and peak separation for each hydrogen isotope gas, and the results will be presented.

This Raman system will be installed at UNITY-2, under construction in Chalk River, Ontario, Canada, where integrated demonstration tests will be conducted in a fusion fuel recovery and injection system using tritium gas.

IDAES/Pyomo-based simulation framework for hydrogen recovery via palladium membranes

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Recovering hydrogen isotopes (H2, D2, T2) as a group from helium-dominated fusion exhaust and blanket purge streams is a key fuel-cycle purification requirement. Dense palladium (Pd)-based membranes suit this task well, permeating hydrogen almost exclusively while rejecting helium and other non-hydrogen species regardless of isotope. Because H2, D2, and T2 nonetheless permeate at measurably different rates (the isotope effect), accurate performance prediction still requires resolving isotope-specific, multicomponent transport. The resulting governing equations are numerically stiff and prone to solver failure. This work presents a physics-based process simulation framework for Pd-membrane hydrogen isotope purification, built entirely on open-source tools (IDAES and Pyomo). A one-dimensional, membrane unit model is implemented. Isotope-specific permeance follows an Arrhenius temperature dependence, with independently fitted parameters for H2, D2, and T2, while the driving force follows a Sieverts' law square-root pressure dependence. The resulting large-scale nonlinear program is solved robustly in IPOPT, using a multi-stage homotopy/continuation strategy with adaptive step-size bisection. Geometry and operating conditions were calibrated against literature Pd-membrane module designs: 183 Pd/Ag tubes membrane module at 673 K, with feed and permeate pressures of 2 bar and 0.3 bar. Under these conditions, the framework reproduces consistent purification performance. Preliminary single-pass simulations recover more than 96% of the fed D2 and T2 to the permeate gas. The framework further supports sensitivity analysis over feed pressure, length, and temperature. It also supports multi-stage series and parallel cascades, offering a flexible tool for tritium exhaust processing design.

Lithium vaporization behavior of two-phase Li8ZrO6-Li2O sintered materials in 0.1% H2/He

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Li2O has the highest lithium atomic density among oxide materials. However, due to its low chemical stability, it is difficult to use it alone as a tritium breeder. Li8ZrO6 has the second-highest lithium atomic density among oxides and is attracting attention as a promising candidate for tritium breeder. It has been reported that a two-phase sintered compact, in which stable phase grains surround unstable phase grains, stabilizes the entire structure. [1] It has also been reported that a two-phase sintered compact consisting of Li8ZrO6 and Li2O can maintain a higher lithium atom density than the conventional candidate material, Li2TiO3, for three years at temperatures below 770 ℃. [2] These studies suggest that a two-phase sintered compact consisting of Li8ZrO6 and Li2O, which contains a larger amount of Li2O, can achieve a higher lithium atom density below 770 ℃. In this study, we prepare a two-phase composite material (Li8ZrO6–Li2O) with a Li/Zr molar ratio of 12. The samples are annealed for extended periods in a high-temperature reducing atmospheric pressure (0.1% H2/He, 320–900 ℃) that mimics the conditions inside the blanket. The crystal structure stability and the lithium vaporization behavior are systematically investigated at each annealing time.
[1] S. Ogawa,et al.,Fusion Engineering and Design, 98–99 (2015), 1859-1863, 10.1016/j.fusengdes.2015.01.042
[2] S. Kikuchi et al., Fusion Engineering and Design, 193 (2023), 113793.

Optimization of the Experimental System for High-Accuracy Benchmarking of Large-Angle Scattering Cross Sections of Lithium with 14 MeV Neutrons

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

In fusion reactors, large-angle neutron scattering has a significant impact on neutronics calculations, particularly for the reactor blanket. Therefore, benchmarking studies of large-angle scattering cross sections are indispensable. In a previous study, a benchmark experimental system using two shadow bars composed of trapezoidal conical iron blocks was developed to validate large-angle scattering cross sections, and a benchmark experiment for lithium was conducted using hafnium as the activation foil. However, the statistical error was large due to neutrons scattered from walls and surrounding materials. Therefore, further error reduction was required to achieve a high-accuracy benchmark.

In this study, new activation foils were investigated to reduce statistical errors. Candidate activation foils were selected based on isotopic abundance, activation cross sections, threshold energies, and half-lives. The γ-ray count rates expected to be measured with an HPGe detector were calculated using MCNP5, and the statistical errors were evaluated. As a result, indium foil was found to be the most effective in reducing statistical errors, followed by magnesium foil.

In addition, the experimental systems using indium and magnesium foils were improved to further reduce errors. The lithium target was redesigned into a cylindrical shape, and the activation foils were also made cylindrical and thicker to increase the detection efficiency for large-angle scattered neutrons. The target table and shadow bars were replaced with thin-legged structures to decrease background scattering. Furthermore, the diameter of the shadow bar and the thickness of the activation foils were optimized. As a result, the statistical errors were successfully reduced in the systems using indium and magnesium foils. In the optimized experimental system, the statistical errors were estimated to be less than 10%. However, because the threshold energy of indium is below 1 MeV, it is sensitive to low-energy background neutrons. Consequently, background neutrons could not be sufficiently eliminated, and large-angle scattered neutrons could not be clearly isolated when using indium foil. Therefore, this study concludes that magnesium is the most suitable activation foil for performing a high-accuracy benchmark experiment on the large-angle scattering cross section of lithium.

Based on these findings, benchmark experiments using the experimental system established in this study are planned for future work.

Evacuation behavior of Na and Li vapor for the development of fusion vacuum system

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Vacuum pump system for the fusion system is quite important regardless of the confinement style, which must have high efficiency and resistance against tritium contamination and radiation. Additionally, it should work in strong magnetic field, without the release of high-Z element depending on the fusion system, without regular maintenance.
The diffusion pump which usually utilizes oil or mercury vapor to transfer the momentum, can be a candidate when the vapor source is replaced to the light alkali metal (such as Li, Na, or K) which is free from radiation degradation. So far, authors have developed an alkali-metal vapor diffusion pump working with K and Na vapor. Recently, a major improvement for the experimental setup has been completed and the evacuation experiment at higher temperature, enabling Li vapor operation and more effective evacuation by Na vapor than the old setup. Using the new setup connected to a 0.5 m3 vacuum chamber equipped with a mass analyzer, a Pirani gauge, an absolute capacitance gauge and a gas injector line, evacuation rates for several gas species are investigated. Evacuated gas species are H2, D2, He, Ne, CH4, N2, Ar and CO2. For some gas species, continuous gas supply is given to quantify the compression rate. The dependence of the evacuation rate on the heating power and cooling condition is also investigated.

#13309

Development of a New Solid Tritium Breeder Blanket

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Many existing solid breeder blanket concepts employ lithium orthosilicate (Li4SiO4), lithium metatitanate (Li2TiO3) or some combination of the two, for example the KArlsruhe Lithium OrthoSilicate (KALOS). When incorporated into breeder blanket designs these ceramics are typically coupled with a beryllium-based neutron multiplier. However, the scarcity, toxicity, and purity concerns prompt the consideration of an alternative multiplier. The objective of this work is to produce a design for a new solid tritium breeding blanket, minimising or eliminating the use of beryllium, considering all suitable lithium ceramics, and optimising the configuration of the blanket for both an acceptable TBR and the thermomechanical properties of the materials. OpenMC Monte Carlo neutronics and coupled neutronics/MOOSE finite element simulations via Cardinal have been used to parametrically optimise blanket materials and parameters [1 - 3]. In this presentation, we show the results of our neutronics simulations investigating the novel use of lead-based intermetallic compounds as a hybrid structural multiplier material and how this may enable the elimination of a beryllium requirement. Preliminary coupled neutronics and finite element simulations have been conducted to define the thermomechanical performance requirements of the candidate materials in addition to the thermal-hydraulic design parameters of the coolant channel configuration and working fluid selection.

Acknowledgments
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[1] P.K. Romano et al. “OpenMC: A State-of-the-Art Monte Carlo Code for Research and Development”. Annals of Nuclear Energy 82, 90–97, (2015).
[2] C.J. Permann et al. “MOOSE: Enabling massively parallel multi-physics simulation”. In: SoftwareX 11, p. 100430, (2020).
[3] A.J. Novak et al. “Coupled Monte Carlo and Thermal-Fluid Modelling of High Temperature Gas Reactors Using Cardinal”. Annals of Nuclear Energy 177, p. 109310, (2022).

#13313

Tritium extraction systems for lead-lithium alloys in mock-up and power plant-scale breeder blankets – analysis, design, and R&D roadmaps

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The tritium extraction system (TES), which interlinks the breeder blanket and the tritium fuel cycle, plays a key role to ensure tritium self-sufficiency and commercial viability of fusion. Effective and efficient TES are essential for ensuring high recovery rates, minimising inventories and losses, and complying with regulatory limits on tritium releases. As private and public programmes progress toward power plant-scale designs, an integrated understanding of TES behaviour across scales and development, from laboratories, test loops, and power-plant-scale systems is increasingly necessary.

This work seeks to provide a framework for the R&D of TES for a diverse range of PbLi-based blankets and blanket mock-ups with the aim of maximising the applicability of existing experimental data and the upcoming test rigs under development. The work consist of the following: (1) analysis of the fundamental transport phenomena governing tritium behaviour in flowing Pb-Li and across interfaces, (2) evaluation of the candidate extraction technologies across relevant and diverse operational regimes, (3) proposal of design requirements for scaled TES integration into various breeder blanket loops, and (4) outline a coordinated R&D roadmap connecting small-scale experimentation with full plant-level deployment taking into account time constraints and risk management.

Key results to be presented include: (1) comparisons between leading extraction technologies focused on power-plant performance and lessons learned from flow loop experiments and how these two approaches can be compatible with each other. (2) how differences in various PbLi-based blanket concepts (e.g. flow velocity, temperature, composition, structural materials) affect the potential design and operation of power plant-scale TES and how to make small-scale experiments relevant for the whole diverse panorama of PbLi blankets. And (3) how to address expected and unexpected challenges during scale-up of TES without impacting the critical path of power plant development. The tools used include models ranging from system to subcomponent scales and TRL and risk analysis frameworks.

Experience and Design Implications from the First Cryogenic Operation of Large Custom Cryopumps Manufactured for ITER Torus, Cryostat, and Neutral Beam Pumping Systems.

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Nine large bespoke cryo-sorption pumps have been successfully manufactured for pumping the ITER divertor and cryostat. The first of these, the pre-production cryopump (PPC), was primarily constructed to validate manufacturing methods; however, it is now considered a test pump that may be installed on ITER for initial operations. These pumps were designed with a nominal molecular flow pumping speed of ~100 $m^3 s^{-1}$ and will operate in synchronization to provide continuous high vacuum pumping of the vacuum vessel and cryostat across multiple operational modes.
In addition, one large bespoke cryo-sorption pump has been successfully manufactured for neutral beam (NB) systems. This first NB pump was produced for the Megavolt ITER Injector and Concept Advancement (MITICA) facility and is of very similar design to the pumps planned for ITER heating and diagnostic NB systems. It was designed with a nominal molecular flow pumping speed of ~4700 $m^3 s^{-1}$ for hydrogen. The pump provides the pumping capability for MITICA operation and can also be regarded as a pre-production

unit for the serial production of ITER NB cryopumps.

Both pump types have now undergone initial cryogenic pumping tests at temperatures approaching 4.5 K. The PPC was tested at the ITER cryopump test facility, while the MITICA cryopump was tested at the MITICA facility in Padova. At the ITER facility, testing was first limited to inert gas injection; therefore, initial experiments were conducted using helium and neon. At MITICA, hydrogen injection was possible for the first experiments, but without the design cryogenic supply flow rates and temperatures.

These cryogenic tests have initiated the validation of the assumptions made during the design of both pump types. The PPC demonstrated helium pumping speeds and capacities exceeding original model predictions for helium. The MITICA cryopump exhibited a good hydrogen pumping speed considering non ideal cryogenic supplies and some 4.5 K pumping sections lacking their full shielding. Further cryogenic testing is now planned for both types of pump, with improved conditions expected to bring more complete validation of performance. Subsequent analysis of the results has driven minor design modifications to enhance the hydrogen pumping performance of the NB cryopump for ITER operation.

This paper summarizes the experience gained and lessons learned from the cryogenic testing of the ITER torus and NB cryopumps, outlines design changes to enhance hydrogen pumping performance on ITER NB systems and proposes recommendations for further optimization of cryopumps for future fusion devices.

Lithium isotope analyses for the validation of the ICOMAX lithium enrichment setup

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

For breeding tritium in fusion power plants, most current blanket concepts require large quantities of lithium, enriched in the isotope lithium-6. Depending on the actual blanket design, lithium with 30 - 90% lithium-6 is needed, due to its advantageous reaction properties. As enriched lithium is not commercially available in the large quantities required, a large-scale process for the production of tons of lithium-6 has to be developed. Up to now most lithium enrichment methods are only available on a laboratory scale. Only the COLEX process was applied on a large scale between 1954 and 1963 in the US. This is based on the lithium isotope exchange between lithium amalgam and aqueous lithium hydroxide, which are conducted through a column in a countercurrent process, enabling continuous lithium enrichment. In order to avoid mercury emissions and environmental contamination, as seen back then, ICOMAX uses the same working principle, but is improved by utilizing fully closed process operation, optimized component design, and stringent emission controls.

This paper focuses on lithium isotope analyses to validate the performance of the ICOMAX components, amalgam maker, enrichment column and decomposer. The lithium isotope ratio of aqueous solutions can be determined via inductively coupled plasma mass spectrometry (ICP-MS). For the analysis of lithium amalgam, the amalgam has to be solved in diluted nitric acid first. After calibration with appropriate lithium isotope standards, the respective samples can be analyzed regarding their lithium-6 and lithium-7 concentrations and respective isotope ratios.

Status and activities of the development of a Metal Foil Pump for fuel separation in a fusion plant fuel cycle

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Independently on the physics principle of a future fusion reactor, a fuel cycle including Direct Internal Recycling (DIR) is indispensable. Herewith, the separation of unburnt fuel from the pumped exhaust gas stream is essential and can be realised by the Metal Foil Pump close to the divertor. This device will, under low pressure, separate all types of hydrogen isotopes sharply from all other gases which the exhaust gas will contain. The separated hydrogen isotopes can consequentially directly be recycled to the matter injection system to fuel the machine. The Metal Foil Pump is based on the principle of superpermeation, which allows hydrogen at low pressures, with the energetic support of a cold plasma, to pass a thin metal foil even against a pressure gradient.
Experimental work with different lab set-ups has been done over years at KIT, accompanied with simulation effort regarding performance understanding and prediction and to find an optimal MFP configuration in a fusion machine. Furthermore, manufacturing aspects have been investigated already with a dedicated set-up.
All these results, lessons learned and experience are now combined for the upcoming next step, the demonstration of an MFP prototype of application-relevant scale. For this purpose, the Karlsruhe Institute of Technology (KIT) and Kyoto Fusioneering Europe GmbH (KFEU) are jointly developing a demonstration unit for a tritium-compatible MFP. This demonstrator will first be tested in the JASON facility at KIT with protium and deuterium and subsequently at UNITY-2 under tritium. This will mark a significant step for the necessary TRL maturation of the MFP.
This contribution presents a comprehensive overview of the MFP development time line, the status and current results of the third-generation experimental set-up at KIT and the outcome of the design development of an MFP demonstrator done within a collaboration between KIT and KFEU. Finally, an outlook with manufacturing and operation aspects will be given.

#13342

Numerical computation of MHD pressure drop for stellarator breeding blanket configurations

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Stellarators are a promising candidate for the first generation of fusion power plants. However, their intricate geometry is challenging from the technological point of view. Breeding blanket (BB) design is not exempt from this difficulty. In liquid metal breeding blanket concepts, magnetohydrodynamic (MHD) pressure drop must be minimized in order to limit its toll on the balance of plant. Pressure drop analysis is done relying on numerical codes and empirical correlations, yet the highly non-uniform magnetic field characteristic of this kind of stellarator configurations pushes them to their limits of applicability.

In this work, we test the numerical tool GridapMHD for stellarator-relevant conditions. GridapMHD is a finite-element-method implementation of a charge-conservative formulation of the inductionless MHD problem. The code's validation and verification is an on-going effort, but it has already provided highly accurate results. Here we aim to extend its validation to the high-Hartmann-number highly-non-uniform magnetic field scenarios expected in stellarators. In order to do so, we compare our code simulations with the well-known results produced in the facility ALEX in 1987. This work provides empirical results of rapidly varying magnetic fields, hence it is a useful validation case for our purpose. After this step, we use GridapMHD to perform an MHD analysis of the state-of-the-art stellarator-based DCLL BB design our team develops. This BB design features an advanced strategy for MHD pressure drop mitigation — aligning the lead-lithium flow with the magnetic field lines by arranging blanket modules in a quasi-toroidal segmentation. In this analysis, we simulate some simplified version of reference channels of this BB configuration using the magnetic field expected along them. Furthermore, we employ the pressure drop results of this and similar analyses to test Miyazaki's pressure drop correlation under stellarator relevant conditions.

Keywords: GridapMHD, MHD pressure drop, stellarator, liquid metal, breeding blanket, code validation.

Acknowledgments
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Preliminary Experimental Study on Hydrogen Oxidation and Regeneration Characteristics of a CuO-Based Oxidizer for Coolant Purification System of Breeding Blanket

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Keywords: Coolant Purification System (CPS), Oxidizer, Oxidation, Regeneration, Breeding Blanket

A CuO-based oxidizer is a key component of the Coolant Purification System (CPS) in a breeding blanket, where it oxidizes hydrogen isotopes and contributes to maintaining the chemical composition of the coolant gas. Since multiple oxidizer beds operate alternately between oxidation and regeneration modes at elevated temperatures, maintaining appropriate thermal conditions is essential to ensure high purification efficiency in the CPS.
To investigate temperature optimization under representative operating conditions, a preliminary experimental study was conducted using the RAVAD oxidizer module installed at KFE. Oxidation and regeneration characteristics were examined under 250–400°C with scaled-down flow conditions [1]. The oxidized and reduced samples were analyzed by X-ray diffraction (XRD) to identify phase transitions among Cu, Cu_2O, and CuO, and to evaluate oxidation and reduction behavior.
The results provide fundamental data for defining the thermally stable operating range for reversible transitions among Cu/CuO/Cu2O in CPS operation for breeding blanket systems. These findings will support the development of a design and kinetic model for oxidizer-integrated purification processes, and further validation will be carried out through extended cyclic experiments.

Acknowledgments
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[1] C. W. Shin et al., Fusion Engineering and Design, Volume 192, 2023, 113716

Progress towards high-fidelity multi-physics simulation of fusion breeder blankets

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The design and analysis of fusion power plants is hindered by a complex interplay of phenomena across multiple physics domains (plasma physics, neutronics, thermal-hydraulics, electromagnetism, material science and structural mechanics) and many orders of magnitude of spatial and temporal scales. We outline a multi-physics, multi-scale modelling approach designed to provide a more holistic and predictive understanding of fusion power plant systems and the progress made towards the objective of physics complete simulation of these systems.

Our approach utilises an architecture that is modular, composable and scalable; capable of linking specialised codes for different physics domains into the core finite-element framework. For the core of the framework we select the Multi-physics Object-Oriented Simulation Environment (MOOSE), which has ample built-in physics and native multi-physics and multi-scale coupling. For specialist physics fields we directly couple external specialised solvers: for neutron transport the framework is coupled to OpenMC using the Cardinal MOOSE application, and for fluid dynamics we couple to either NekRS or OpenFOAM depending on the required fidelity and whether the computational hardware is GPU or CPU. We extend the fluid dynamics capability to include magnetohydrodynamics, enabling the large-scale simulation of liquid-metal flows influenced by the strong magnetic fields found in magnetic confinement fusion devices. We also enhance the heat transfer modelling with multi-phase, boiling flows to enable capability for analysis of advanced thermal devices such as hypervapotrons.

With this highly-coupled multi-physics toolchain, we demonstrate its application to the detailed simulation of fusion power plant systems. The breeder blanket environment provides an excellent test for the framework covering the full spectrum of engineering physics. The successful implementation of this framework promises to accelerate the development cycle of fusion power plants. It enables high-fidelity virtual prototyping, allowing engineers to test and optimise designs before physical construction, reducing both costs and risks associated with power plant design. This complexity encapsulating approach has the potential to reveal emergent physics interactions which could lead to black swan events.

Investigation of a Microwave Plasma-Driven Metal Foil Pump for Hydrogen Separation in Fusion Fuel Cycles

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The separation of unburnt deuterium and tritium from exhaust gases and their subsequent recirculation into the plasma chamber is a key element of the fuel cycle in future fusion power plants and is essential for their reliable operation. One promising approach for selective hydrogen isotope separation is the metal foil pump (MFP), which combines a hydrogen-permeable metal foil with a plasma source. Driven by the principle of superpermeation, hydrogen isotopes can permeate through the foil selectively, even against an adverse pressure gradient.

This work investigates a plasma-driven metal foil pump with a focus on the detailed characterization of the plasma and its interaction with surrounding components. A linear microwave plasma source, the Duo-Plasmaline, is coupled to a hydrogen-permeable niobium foil within the new flexible experimental setup HYPERPLEX (HYdrogen PERmeation PLAsma EXperiment). Particular attention is given to the mutual interaction between the plasma source, the metal foil, and the vacuum chamber, and their influence on plasma properties and stability.

Plasma characterization is performed over a wide range of operational parameters using a comprehensive set of diagnostic techniques. Optical emission spectroscopy (OES) and Langmuir probe measurements are employed to determine plasma composition, electron density, and electron temperature, while energy-dispersive mass spectrometry (EDMS) is used for residual gas analysis and to measure ion species and ion energy distributions. The influence of operational parameters on plasma characteristics is systematically investigated and correlated with hydrogen permeation through the metal foil.

A key aspect of this study is the measurement and optimization of the hydrogen permeation flux with respect to relevant plasma parameters. In this context, plasma impurities are identified and quantified, and their dependence on plasma–surface interactions within the experimental setup is investigated. The impact of these impurities on plasma behavior, ion energies, hydrogen permeation efficiency, and surface modification of the niobium foil is analyzed. Furthermore, long-term plasma operation is examined to assess stability, aging effects, and potential degradation mechanisms. The aim is to provide a detailed understanding of the interplay between plasma characteristics and hydrogen permeation in plasma-driven metal foil pumps and support their assessment for application in future fusion fuel cycles.

First Results from the Novel ‘ROBUST’ Approach to Integrated Breeding Blanket Design

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The breeding blanket is a highly integrated part of a fusion plant, and has the critical functions of tritium breeding and power extraction whilst maintaining structural integrity under thermal and electromagnetic loads, plasma disruption events and high neutron fluences. Despite decades of R&D, blanket technologies remain relatively immature and are unlikely to be tested in such an integrated environment within the next decade, presenting a substantial risk for fusion plant design programmes. Consequently, it is crucial that blanket engineering design is approached using a methodology that maximises the use of digital system engineering tools, to provide a structured and logical selection of breeder blanket technologies and design attributes based on the specific constraints imposed by a given reactor. While traditional engineering tools (FEA, CFD etc.) are commonly applied to blanket designs, this is often in isolation of the full fusion plant context, and there is currently low adoption of integrated design and analysis tools linked to system requirements. Such tools are common in more well-established industries (e.g. Aerospace, Automotive) and the fusion community would benefit from similar toolsets, to manage the huge complexity of a fusion plant, allow traceable decision making, and to align to industry standards. These tools would be able to sweep the available design space and quantitively assess 100-1000s of blanket concepts rapidly, to provide a shortlist of blanket concepts that are deemed the most viable.

This work has developed an integrated system engineering design and analysis framework for fusion blankets, named ROBUST, which determines the most viable blanket conceptual designs for a given set of reactor requirements. The process first establishes the problem definition space, utilising industry-standard systems engineering methodologies. These include requirements capture, requirements validation and verification, Model-Based Systems Engineering (MBSE), Failure Modes & Effects Analysis (FMEA) and by employing a structured material down-selection process. Then, to generate and assess blanket designs relative to the problem definition space, a multiphysics analysis workflow has been built using Ansys OptiSLang, which provides quantitative assessment of blanket concepts against key performance criteria. The pre-conceptual systems analysis workflow encompasses neutronics, thermal hydraulics, structural analysis and fuel cycle assessments to assess design concepts.

Both the systems engineering processes and the analysis workflow have been applied to a spherical tokamak use case, which provides an example set of reactor constraints. This work presents the findings of this study and a roadmap to developing a progressively more advanced capability in the future.

De-risking Glovebox Delivery for the H3AT Tritium Fuel Cycle

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The UKAEA-Eni Hydrogen-3 Advanced Technology (H3AT) Tritium Loop facility is a closed-loop tritium fuel cycle pilot plant, developed to support fusion technology research and development. The facility comprises multiple interdependent fuel cycle subsystems housed within gloveboxes, which form the core of the containment and operational architecture. Gloveboxes are a critical enabling technology for fusion fuel cycle facilities, providing a controlled environment for the safe handling of tritium while accommodating complex process equipment. Their design must reconcile confinement, tritium compatibility, operability, maintainability, and integration within highly constrained space and weight envelopes.

The fuel cycle includes 27 gloveboxes housing varying process functions, to be delivered on a demanding programme timeline. Managing technical, manufacturing, and programme risk at this scale presents a significant challenge. The formation of sub-assemblies that are common to groups of gloveboxes will be essential to allow for efficiencies during fabrication. Additionally, these sub-assemblies will standardise operation and maintenance throughout the life of the gloveboxes, allowing for shared parts and consumables.

An early glovebox design strategy has been adopted, in which two early prototype gloveboxes are designed and fabricated ahead of the remaining suite to reduce risk across the wider programme. This approach enables early resolution of key challenges, including penetration strategy, thermal management, space and weight constraints, confinement philosophy, over- and under-pressure protection system integration, and the accommodation of fuel cycle processes. The selected prototype gloveboxes each present distinct packaging and supporting system challenges, allowing for de-risking of future gloveboxes and providing an avenue for iterative design learnings from early testing. This strategy allows for practical lessons to be learned, without introducing significant rework across the entire glovebox fleet.

This approach places particular emphasis on how process design decisions directly inform mechanical glovebox design, enabling more effective system integration. The strategy will allow risks to be mitigated on a small number of gloveboxes while allowing design to mature and scale to a larger throughput. The lessons learned from this approach are expected to support more efficient alignment of design and fabrication activities and provide a transferable framework for glovebox delivery in future fusion fuel cycle facilities.

#13415

Tritium behaviour in thermal insulation materials and assemblies

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

A coordinated experimental programme was carried out to characterise tritium behaviour in materials and assemblies relevant for process lines with tritiated helium under high pressure and high temperature conditions, like parts of ITER Test Blanket Systems*. Firstly, the basic HT and HTO transport, sorption, and desorption properties of silica based microporous insulation (Microtherm®) and selected passive fire protection materials were established through loading experiments in a temperature range from RT to 800 K, subsequent permeation and diffusion measurements, complemented with thermal desorption spectroscopy, and microstructural and chemical characterisation. HT transport was found to be dominated by gas transport through open porosity, while HTO behaviour is governed by water related interactions, leading to higher uptake at low temperature and decreasing solubility with increasing temperature. Most absorbed tritium is released at moderate temperatures, with only a limited residual fraction remaining.
Secondly, the effects of repetitive loading and long term exposure were investigated for few Microtherm® grades. Repeated HTO loading–outgassing cycles and long term HT exposure up to 200 days showed no ageing or memory effects. Tritium uptake and release behaviour remained stable, with a constant residual fraction of approximately 10–20% that did not increase with cycle number or exposure duration.
Thirdly, the HT permeation and retention in Type 316L stainless steel tube assemblies were studied, both with and without thermal insulation, under representative scaling (DN25 x 300 mm permeation section) and operation conditions (Helium + 0.1% H2 and HT gas loading, tube temperatures up to 773 K, permeation chamber with air). Tritium permeating through the stainless steel was recovered predominantly as HTO, and measured permeation strongly depended on exposure history and system conditioning. A significant fraction of the tritium inventory was only recovered during post test baking, demonstrating that permeation measurements alone may underestimate retained tritium. In insulated assemblies, the largest inventories were associated with metal components, while insulation accumulated a smaller but measurable HTO dominated inventory.
Together, the results provide a consistent experimental basis to elaborate further on system engineering design and scoping performance assessments, including both chronical contamination and transient tritium outgassing impact on normal, maintenance and dismantling operations.
The paper will describe the experimental set-up and results obtained for the materials and assembly behaviour and refer to the tritium transport modelling using COMSOL Multiphysics as applied for experiment design and interpretation.

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#13429

Monolithic Spatial Heterodyne Spectroscopy as a Robust Tool for Tritium Accountancy

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Fusion energy will provide a clean energy source well-suited for on-demand dispatchable electricity production, supplementing intermittent renewables and fission. The DT fusion fuel cycle is a complex ensemble of components dedicated to processing, extracting, and storing tritium, which means that tritium will constantly be flowing through a commercial fusion pilot plant. As tritium is radioactive, it needs to be accurately accounted for inside the fuel cycle to monitor inventory in different parts of the system as well as transport into and out of the plant. This inventory needs to be accounted for across key systems to ensure continuous operations, allow for shut down, start-up or other off-normal operations, determine if inventory needs to be reduced or augmented through tritium transfers with offsite storage, or meet other operational requirements. Creation of cost-effective accountancy techniques is necessary and an important aspect for the success and efficient operation of fusion reactors. This requires high-sensitivity and high-speed detection solutions that can operate in challenging environments. A spectroscopic technology that can be applied to tritium accountancy in extreme environments is the spatial heterodyne spectrometer (SHS). Spatial heterodyne spectrometry is capable of high spectral resolution, large spectral range, and very high light throughput. This technology has been deployed by NASA as a rugged, radiation resistant spectrometer for space exploration. The system design lends itself to miniaturization and the use of monolithic construction techniques, which can be an advantage when spectrometer size and robustness are important. This research aims to develop a monolithic spatial heterodyne spectrometer (mSHS) that can survive in high flux neutron environments and exposure to tritium to advance diagnostic capabilities for tritium accountancy. The successful development of a compact and robust spectrometer for real-time tritium accountancy will be critical to the success and operation of commercial Fusion Energy.

#13472

Effect of Hydrogen Purging on Tritium Release Behavior from Neutron-irradiated FLiBe Molten Salt

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

FLiBe (LiF-BeF2) is considered a primary candidate for liquid breeder blankets in fusion reactors due to its high tritium breeding ratio and excellent thermal-hydraulic properties. The low solubility of hydrogen isotopes in FLiBe results in a high partial pressure of the tritium bred in the molten salt, which is expected to effectively recover the bred tritium by gas-liquid contact systems. To design an adequate recovery system for the bred tritium, it is also crucial to understand the chemical forms of tritium released from the molten salt into the gas phase. This study investigated tritium release rates and chemical forms from neutron-irradiated FLiBe using a cold trap and water bubblers with a CuO catalyst.

Solid FLiBe samples were irradiated at the Kyoto University Research Reactor (total fluence: 1.7×1015 cm-2). The tritium release experiments were subsequently performed at Kyushu University. Each irradiated sample was placed in a Mo crucible within a quartz tube. Using pure Ar purging at 600 °C as a baseline, the effect of introducing a 1% H2 atmosphere on tritium release behavior was investigated. A cold trap collected tritiated water vapor (T2O), while a dual-bubbler system separated TF and T2, the latter being oxidized via a CuO bed at 400 °C. Tritium concentrations were measured every 10 minutes using a liquid scintillation counter.

The release of both TF and T2 was observed to initiate near the melting point of FLiBe (approx. 459oC). The initial release of T2 was attributed to the reduction of TF through reactions with metallic components in the experimental system. Using pure Ar carrier gas as a baseline, the ratio of chemical forms was determined to be 85.7% of T2O, 11.7% of TF, and 2.6% of T2. This high fraction of T2O is potentially attributed to the presence of BeO impurities within the FLiBe. To further evaluate the influence of BeO impurities on the chemical forms, we plan to conduct comparative experiments using FLiBe samples subjected to HF treatment. Notably, comparative experiments showed that the addition of H2 to the purge gas significantly enhanced the tritium recovery rate. It is hypothesized that a portion of tritium may remain adsorbed on the melt surface or trapped within the salt under pure Ar, and the introduction of H2 effectively clears this residual tritium by promoting isotope exchange reactions. This indicates that the composition of the purging gas has a significant impact on tritium release kinetics.

Pneumatic Capsule Transportation Tests Conducted by WCCB-NAS Mock-up Test Facility

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The Water-Cooled Ceramic Breeder Test Blanket System (WCCB-TBS), has been developed by National Institutes for Quantum Science and Technology (QST) to conduct tests in the ITER. It is planned that an irradiation end of TBS Neutron Activation System (NAS) will be built in the Sub-Module (SM) of the WCCB Test Blanket Module (TBM). The WCCB-NAS transports irradiation capsules to the SM and withdraws by pneumatic system to measure neutron flux and fluence. The gross amount of tritium produced by the TBM will be estimated based on the neutron flux and fluence during a plasma operation. For integration with ITER utilities, it is necessary to inform ITER organization of the amount of gas consumed during WCCB-NAS operation must be quantified and reported to the ITER organization. To optimize the WCCB-NAS operation and determine suitable capsule geometry the QST constructed a full scale NAS mock-up test facility. Both conventional dog-bone capsules — used in reactors and fusion facilities — and spherical capsules were investigated because their geometry provides superior transport robustness particularly at discontinuities such as pipe joints or connection sections, where they are less prone to sticking or mechanical interference compared to dog-bone capsules. Spherical capsules also offer greater mechanical robustness against stress concentrations than dog-bone ones. This study evaluated the transportability and gas consumption of candidate capsules under various operating conditions using the mock up. The transportation of spherical capsules was successfully demonstrated for the first time in a full-scale NAS facility. Dog-bone capsule and Φ8.0 mm and 7.5 mm steel balls could be transported by high pressure gas. Higher gas pressure increased transport speed and gas consumption, while still preventing sticking. However, dog-bone capsules occasionally got stuck due to lid displacement during transportation. R&D results in Japan and consideration related to the WCCB-NAS will be presented on the symposium.

#13509

The EU Breeding blanket programme for stellarators: designs, integration and tools

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Inside the rapidly growing stellarator-focused ecosystem, the EUROfusion programme provides a coordinated, publicly driven, physics-based and reactor-oriented framework for breeding blanket development in next-generation stellarators.

Since 2021, CIEMAT, within the EUROfusion framework, has led the comprehensive development of the EU Breeding Blanket (BB) programme for stellarators in the HElical Advanced Stellarator (HELIAS) configuration strongly focused on the Dual Coolant Lead Lithium (DCLL) BB concept. This work has encompassed conceptual design, system integration, and the development of dedicated computational tools, establishing a consistent methodological and technical foundation for breeding blanket studies in stellarator devices.

Building on this experience, the programme is now transitioning to a quasi-isodynamic (QI) stellarator configuration, which is strongly supported by plasma physics results and widely regarded as a promising pathway towards a reactor-relevant stellarator concept. This transition represents a significant step forward, enabling the systematic transfer and adaptation of the expertise, integration methodologies, and modelling capabilities developed for HELIAS to a configuration with clearer reactor perspectives.

The programme introduces important advances in blanket concepts, adding to the DCLL approach other well-grounded solutions such as the Helium-Cooled Pebble Bed (HCPB) BB. In parallel, substantial progress is being made in advanced modelling and multi-physics techniques for magneto-hydro-dynamics (MHD), neutronics, thermal–hydraulics and thermo-mechanical analyses incorporating state-of-the-art computational methods for improved evaluation of pressure drop, mass flow rates, velocities, tritium breeding performance, nuclear heating, shielding effectiveness, material damage, deformation and structural behaviour (among

others) in complex 3D devices with strong inhomogeneity.

In addition, system-level integration activities are playing a decisive role in shaping BB development. Dedicated remote handling strategies tailored to stellarator geometrical constraints, improvements in coil design to enhance maintainability and spatial allocation, and novel First Wall solutions are being pursued in close coordination with blanket design. These interfaces are determinant for overall reactor feasibility, as they directly impact accessibility, shielding requirements, thermal performance, and structural continuity.

Altogether, this contribution presents a coherent and forward-looking framework for breeding blanket development in next-generation stellarators, integrating physics-based design criteria, engineering interfaces, system-level considerations, and enhanced multi-physics analysis and modelling capabilities to support the advancement of stellarator reactor concepts.

Exploring Lithium-Rich Ceramics for Tritium Breeding: From Established to Novel Materials

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The successful implementation of nuclear fusion relies on the achievement of tritium self-sufficiency within the reactor fuel cycle. As tritium is scarce in nature, it must be bred in situ in the breeding blanket through neutron interactions with lithium, predominantly via the $^{6}Li(n,\alpha)^{3}H$ reaction. Lithium-rich ceramic materials are of exceptional interest due to their elevated tritium breeding ratios resulting from a high lithium atom density. However, the properties, stability and compatibility of such materials with structural components, especially under practical conditions, have not been adequately evaluated thus far.

In this study, the synthesis and characterisation of lithium-rich ceramic breeder materials is conducted, focusing on octalithium zirconate (Li_8ZrO_6) as well as a proposed novel compound for tritium breeding. A solid-state synthesis route under inert atmosphere was chosen. Homogenisation was achieved by ball milling in isopropyl alcohol, followed by die pressing, calcination and sintering to obtain dense ceramics. Additional synthesis tests of Li_8ZrO_6 production were conducted in air to assess possible low-cost processing routes compatible with large-scale fabrication constraints.

The fabricated ceramics were characterised by powder X-ray diffractometry (XRD) with regard to their phase composition and scanning electron microscopy coupled with energy-dispersive X-ray spectroscopy (SEM-EDX) to analyse their microstructure and elemental distribution. Thermal properties and phase stability were evaluated using thermogravimetric analysis and differential scanning calorimetry (TG/DSC).

The results indicate that the lithium-rich ceramics can be adequately produced under the chosen experimental conditions. The sintering of the respective compound yields ceramics suitable for further material development and testing. The findings of this study support the continued development of lithium-rich ceramics, and identify a promising new candidate meriting further investigation with respect to its potential application as a tritium breeder ceramic.

Validation of an OpenFOAM solver for liquid metal magnetoconvection: Raylegh-Bénard with an applied vertical magnetic field.

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Liquid metal (LM) blankets in fusion reactors operate under strong magnetic fields and play a key role in heat extraction for energy conversion. In blanket concepts where liquid metals primarily serve as tritium breeders and are circulated at low velocities to minimise MHD pressure losses, reduced inertial forces allow buoyancy to dominate the force balance. Under these conditions, electromagnetic forces strongly interact with natural convection, modifying flow structures and directly influencing the thermal–hydraulic performance of LM blankets. Accurate numerical prediction of magnetoconvective effects is therefore essential for the reliable design, optimization, and safety assessment of liquid metal blanket systems.

This work investigates the interplay of buoyancy, viscous, and electromagnetic forces and their combined influence on convective heat transfer in a liquid metal layer confined between two isothermal horizontal boundaries assumed to be perfectly electrically conducting. The layer is heated from below and cooled from above, and subjected to a uniform vertical magnetic field. A numerical solver is developed to simulate Rayleigh–Bénard convection and validated against experimental Nusselt numbers reported by Burr and Müller [1] for a low-Prandtl-number (Pr ≈ 0.02) sodium-potassium alloy (Na^22 K^78). The experimental configuration consisted of a rectangular liquid-metal layer with dimensions 20 mm × 200 mm × 400 mm, covering Rayleigh numbers in the range 10^3 < Ra < 10^5 and Chandrasekhar numbers up to Q = 1.44 × 10^4. To reduce the high computational cost of resolving thin MHD boundary layers over large domains, several reduced computational domains are explored: 20 mm × 200 mm × 40 mm with cyclic boundary conditions, 20 mm × 80 mm × 40 mm, and 20 mm × 50 mm × 50 mm. Comparison of the resulting Nusselt numbers with experimental data identifies an optimal computational configuration that preserves heat transfer accuracy while significantly reducing mesh requirements.

The Nusselt numbers calculated under the Boussinesq approximation and using a fully temperature-dependent property model are compared to isolate the effect of each modelling assumption on the predicted heat transfer. The differences between these cases indicate: (i) the sensitivity of the results to non-Boussinesq density variations in the buoyancy term, and (ii) the additional impact of allowing viscosity, electrical conductivity, thermal conductivity, and heat capacity to vary with temperature, relative to assuming constant properties evaluated at the mean system temperature.

Keywords: Liquid metal (LM) blankets, Heat transfer, MHD, Rayleigh–Bénard convection, Fusion reactors.

Fabrication and neutron irradiation of Pb2Li7 and Li-6 enriched Pb15.7%Li specimens

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

A single gigawatt-scale fusion reactor will burn ~56 kg tritium per year while the current worldwide commercial tritium inventory is only ~25 kg. Consequently, efficient tritium breeding must be an integral part of a fusion reactor's fuel cycle. Tritium breeder materials all contain lithium as the primary tritium producer and are classified as solid or liquid. Liquid breeders include lead-lithium eutectic (LLE, PbLi with 15.7 at% Li), FLiBe, or Li. LLE has long been considered a favorable liquid breeder material due to its low melting temperature of 235°C, relative chemical safety, and inherent neutron multiplier. Potential solid breeders are almost all lithium ceramics. One of the few metallic potential solid breeders is Pb2Li7, which, like all materials, has advantages and disadvantages. Pb2Li7 has a high lithium density, an inherent neutron multiplier element, high theoretical tritium productivity, and favorable tritium release characteristics. The primary disadvantage is its low melting temperature (726°C) relative to other potential solid breeder ceramics.

Two fission irradiation campaigns are being conducted to irradiate 1) LLE with Li-6 enrichments of natural, 25, 50, 75, and 95.5%, and 2) Pb2Li7 alongside fourteen other potential solid breeder samples. Li-6 for these experiments comes from legacy stockpiles. The fabrication methods for small batches of samples, especially when utilizing Li-6, differ from large scale production methods. The methods used to fabricate specimens are described, where LLE is fabricated using a melt, casting, cutting and pressing process, and Pb2Li7 is fabricated grinding Li granules into Pb powder at stoichiometric levels, and pressing into a pellet. The objective of the PbLi eutectic irradiation campaign is to quantify tritium production as a function of Li-6 enrichment in a well-characterized neutron spectrum and validate or refine neutron cross sections. The objective of the Pb2Li7 irradiation is part of a solid breeder campaign seeking to irradiate novel solid breeder candidates, reduce legacy uncertainties, and develop a process to facilitate rapid neutron irradiations. The fabrication methods and initial irradiation observations are reported herein.

STEP FUSION POWERPLANT BREEDER BLANKET DESIGN APPROACH FOR TRITIUM SELF-SUFFICIENCY

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The breeder blanket for the STEP Fusion Powerplant must provide high performance breeding for a spherical tokamak without inboard breeding, materials and coolant compatible with a 600 °C outlet temperature for net power confidence, and a system deliverable on the targeted timescales of the STEP Fusion programme. Following a comprehensive assessment of all breeder, coolant, and structural material options, solid ceramic lithium oxide (Li2O) has been selected together with a Ti-modified austenitic stainless steel structural material, CO2 coolant, and beryllium-based multiplier.

However, engineering realisation of a deployable blanket on STEP's timescales still requires rapid progress in design alongside fail-fast testing and technology demonstration. This work builds upon initial scoping of the design space to provide an updated view of the STEP Fusion powerplant breeder blanket segment internal design, and wider integration within the tokamak; including an update on the technology demonstration pathway.

For the chosen palette of materials, identifying a performant architecture presents initial design challenges around achieving sufficient confidence in tritium breeding performance, respecting material temperature limits through robust heat management and hydraulic design, and ensuring compliance against availability and reliability requirements. Developing the breeder blanket design which is cognisant of nearby spatial challenges presents additional design constraints.

Meanwhile, use of Li2O as a breeder material has been well documented in literature to present challenges with material degradation, most frequently citing irradiation swelling, LiOH formation, and structural material compatibility. However, these issues have a strong dependence on temperature, environment, and operational duty cycle. The issues have therefore been reviewed from first principles and revisited in the context of the requirements. From this, specific design constraints and opportunities have been identified that further refine understanding of the Li2O breeder blanket design space and feed back into the design process.

Remaining uncertainties and risks have led to a set of prioritised steps for testing and technology demonstration. Preliminary screening of the extent of degradation mechanisms (and sensitivity to operationally controllable parameters) can be carried out in unirradiated environment, before more costly and time-intensive irradiation tests are required. In the longer term, scale-up towards component-level functional testing is required, aiming for proof of mechanical, thermal, and electromagnetic performance

demonstration, in parallel with nuclear and tritium transport performance demonstration. A timeline for this suite of development needs will be presented to give an overall outlook for development of the Li2O blanket concept, including an update on recent developments.

SHIELD: A Gas Driven Permeation rig for Hydrogen Permeation Barrier testing

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The SHIELD project presents a newly developed Gas-Driven Permeation (GDP) rig for measuring hydrogen permeability through structural materials and candidate hydrogen permeation barrier coatings under controlled pressure and temperature conditions. In operation, hydrogen is introduced into the upstream volume at a defined static pressure, while the downstream volume is initially evacuated. Hydrogen permeation through the sample results in a measurable pressure rise in the downstream section, from which hydrogen permeability is determined from the temporal evolution of the pressure signal.

The platform enables independent control of upstream and downstream environments and combines complementary pressure diagnostics to characterise both vacuum quality and hydrogen permeation. Each section of the rig is equipped with full-range Pirani and cold-cathode gauges to monitor pump-down performance, leak-tightness, and background pressure, while high-accuracy capacitance manometers (MKS Baratron unheated 626D) measure hydrogen pressure during permeation experiments. Each capacitance manometer has a resolution of 0.05% of full scale, corresponding to a minimum detectable pressure of approximately 5×10^{-4} Torr in the downstream. Prior to gas exposure, both the upstream and downstream sections are evacuated to high vacuum, typically reaching pressures below 10^{-6} Torr.

Samples consist of 19 mm diameter metallic coupons mounted in high-vacuum-compatible metal-face-seal VCR fittings and are heated in a split-tube furnace, with a thermocouple in contact with the sample to ensure consistent sample temperature. Initial commissioning experiments have focused on uncoated stainless steel to establish baseline permeability and validate repeatability. Following commissioning, SHIELD is now used to measure the hydrogen permeability of samples with tungsten, silicon carbide, and alumina coatings deposited via physical vapour deposition and plasma spraying. Comparative permeation measurements are used to assess coating performance and permeation-reduction factors relevant to fusion and hydrogen-handling applications.

#13538

Deuterium permeation and corrosion behaviors of enamel-coated steel with ceramic breeder pebbles

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Tritium permeation barriers have developed for nearly half a century to mitigate fuel loss and radiological hazards in fusion reactor blankets. Recently, research on ceramic coatings has shifted its focus to address corrosion by tritium breeders and the effects of irradiation. Exposure to ceramic breeder pebbles under blanket conditions caused not only corrosion of the ceramic coating surface by chemical reactions but also cracking and delamination by the growth of an iron-chromium oxide on a reduced activation ferritic/martensitic (RAFM) steel substrate because of oxygen diffusion through the coating. To mitigate oxygen diffusion to the substrate, silicon dioxide-based enamel coatings were prepared and showed an effective reduction in deuterium permeation. In this study, we performed exposure tests to solid breeder pebbles and deuterium permeation tests on enamel coatings under exposure to the solid breeder pebbles to investigate corrosion behavior and its effect on deuterium permeation.

Two types of enamel coatings were fabricated using a ready-made enamel liquid and a custom-made enamel liquid provided by Nihon Horo Yuyaku Co., Ltd., Japan. The RAFM steel F82H plate was dipped in the enamel liquid, withdrawn at a constant speed of 1.0 mm/s, dried at 423 K for 6 min, and heated at 823 K for 2 min in air. Thereafter, the sample was heated at 1023 K for 10 min under a flow of a mixed gas of argon and hydrogen. The coated samples were exposed to ceramic breeder pebbles of lithium metatitanate and lithium orthosilicate at 823 K for 2, 16, and 32 days under argon with 0.1 vol% hydrogen gas flow. Gas-driven deuterium permeation measurements were performed for the coated samples at the driving pressure of 20–120 Pa and 573–823 K under exposure to the ceramic breeder pebbles. Characterization of the samples before and after the permeation measurements was conducted by scanning electron microscopy with energy-dispersive X-ray spectroscopy and X-ray diffraction.

Deuterium permeation measurements showed that the permeation flux decreased by 1–2 orders of magnitude at the first measurement at 673 K in comparison with the sample without exposure to the pebbles. After the measurements at up to 823 K, the permeation flux decreased by 3–4 orders of magnitude at the second measurement at 673 K, suggesting the modification of crystallinity of the coatings during the measurements without degradation. These results also indicate that oxygen diffusion was small to form the iron-chromium oxide of the RAFM substrate.

#13540

Tritium containment property within alumina container for high temperature gas cooled reactor with tritium production capability

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

In order to start up an initial fusion reactor and conduct technical tests on related tritium circulation systems, an adequate amount of tritium (T) must be prepared. To this end, the production of tritium using a high-temperature gas-cooled reactor (HTGR) has been proposed [1]. It has been reported that approximately 600 g of tritium can be produced in one year of operation with a 600 MWt HTGR [2]. At this stage, our aim is to produce tritium compatible with electricity production and heat sources for hydrogen production using an HTGR. We intend to use zirconium (Zr) spheres as an absorption material to prevent an increase in the inner tritium (T) partial pressure in the lithium (Li)-loading rod during this operation, thereby reducing T outflow. We have demonstrated that stable absorption performance can be expected from Zr spheres with a Ni coating at temperatures of 900°C, even when Zr coexists with oxides (e.g. $LiAlO_2$ and Al_2O_3) in the Li-loading rod [2].

We conducted an irradiation test on the Al_2O_3 container with a volume approximately 1/2000 of that of a real HTGR, using the JRR-3 research fission reactor at JAEA. We measured the amount of T that flowed out of the Al_2O_3 container and the amount released from the $LiAlO_2$, comparing the results with those of the simulation. Details of the experiments and a discussion of the Al_2O_3 container's T containment properties will be presented at the conference.

[1] H. Matsuura, et al., Nucl. Eng. Des. 243 (2012) 95.
[2] H. Matsuura, et al., Fusion Eng. Des. 197 (2023) 114054, 215 (2025) 114944.

Eddy Dimensionality-Based Turbulence Modeling for Molten Salt Fusion Blankets

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The design of molten salt breeder blankets for fusion reactors requires multiphysics simulation tools able to capture the coupled effects of neutronics, magnetohydrodynamics, and thermal transport. This work presents recent developments of the nemoFoam solver aimed at improving turbulence modeling for MHD flows in high-Prandtl-number fluids, beyond previously adopted RANS approaches.

Earlier studies mainly relied on cubic stress strain correlation in the turbulence models with the inclusion of magnetic damping terms derived from characteristic electromagnetic braking time scales. In contrast, the present work explores a structural description of turbulence, based on the evolution of eddy dimensionality. To this end, the k–ε–α three-equation turbulence model proposed by Widlund is implemented. The model was originally developed to describe the transition from fully isotropic, three-dimensional turbulence to quasi-two-dimensional turbulent structures under strong magnetic fields. In the context of fusion blankets, this structural formulation provides an alternative to explicit damping source terms while preserving physical consistency in high-Prandtl molten salt flows.

The numerical framework is implemented within OpenFOAM and applied to ARC-relevant blanket geometries. Comparative analyses investigate how different turbulence closure strategies, including cubic models with electromagnetic damping and eddy dimensionality-based models, affect flow organization and thermal transport predictions. The results aim to clarify the respective roles of turbulence anisotropy and energy dissipation in determining thermal hotspot formation and overall heat transfer performance in molten salt breeder blankets.

#13563

Characterisation of paramagnetic radiation-induced defect centres in two-phase lithium orthosilicate – lithium metatitanate composite material exposed to various types of ionising radiation

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Two-phase lithium orthosilicate – lithium metatitanate composite material is a promising candidate for tritium breeding in future nuclear fusion reactors. The tritium breeding material will operate under extreme conditions, including high fluxes of neutrons and other types of ionising radiation, elevated temperatures, strong magnetic fields, etc. The interaction mechanism of ionising radiation with matter depends on the mass, electric charge, and energy of the incident particles. In the present work, the formation and accumulation of paramagnetic radiation-induced defect centres is investigated in the composite material under exposure to various types of ionising radiation. For irradiation with X-rays, an X-ray source operating at voltage of 45 kV was used. For irradiation with accelerated electrons, the linear accelerator “Elektronika 10/10” was employed. For irradiation with thermal neutrons, a critical experimental reactor facility will be utilised. Afterwards, the irradiated samples were analysed using electron paramagnetic resonance (EPR) spectroscopy. On the basis of the obtained results, it can be concluded that the detected EPR signals with $g \geq 2.00$ mainly originate from oxygen- and silicon-related centres in the lithium orthosilicate phase. In contrast, the signals with $g < 2.00$ are attributed to titanium-related electron centres, arising from the isovalent substitution of silicon by titanium in the crystal structure of the lithium orthosilicate phase. The EPR signals of the lithium metatitanate phase are not detected in this case due to the high radiation stability of this compound in comparison to the lithium orthosilicate phase. Using the isochronal annealing method, it was determined that the annealing of these accumulated paramagnetic radiation-induced defect centres proceeds in multiple stages up to 350 °C.

This work has been carried out within the framework of the Latvian Council of Science project No. lzp-2024/1-0162 “Influence of stoichiometry on radiation-induced effects in advanced two-phase functional materials for future thermonuclear fusion reactors”.

#13578

Features of solid-state synthesis of rare-earth metal beryllides MeBe13 (Ho, Yb, Er) from beryllium powders and their hydrides

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Beryllides of the MeBe12 and MeBe13 types represent a promising class of materials for operation under extreme conditions, including high-temperature and intense radiation loads, characteristic of nuclear and fusion energy systems, such as breeding zones, components of the tritium fuel cycle, and high-temperature heat-removal elements. In this context, rare-earth metal beryllides attract considerable interest as potential radiation-resistant, heat-resistant, and corrosion-resistant materials. However, experimental data on the properties of lanthanide beryllides remain limited, largely due to the technological challenges associated with their synthesis.

In this work, the feasibility of synthesizing MeBe13 beryllides (Me = Ho, Yb, Er) by a solid-state reaction route using mixtures of beryllium powder and corresponding rare-earth metal hydrides of the MeH2 type under vacuum conditions was investigated. The phase composition and morphology of the initial powders were characterized, the processes accompanying the heating of holmium, erbium, and ytterbium hydrides were analyzed, and the temperature–time conditions governing the formation and stability of the MeBe13 phase in the 800–1300 °C range were determined. The thermal behavior of compacted powder mixtures of beryllium and rare-earth metal hydrides with stoichiometry corresponding to MeBe13 was additionally studied by simultaneous TG–DSC–MS analysis up to 1500°C.

A comparative analysis of the Ho–Be, Yb–Be, and Er–Be systems was performed with respect to:

(i) the onset temperature of MeBe13 phase formation and the time required for completion of synthesis,

(ii) the characteristics of rare-earth metal hydride dehydrogenation and subsequent high-temperature solid-state reactions,

(iii) microstructural evolution and porosity development, and

(iv) thermal effects and gas release behavior derived from TG–DSC–MS data.

The results confirm the formation of $MeBe_{13}$ beryllides with a cubic $NaZn_{13}$-type crystal structure and provide experimental insight into the temperature–time windows of their formation under solid-state conditions, as well as into the thermal behavior of compacted mixtures of the initial powders and rare-earth metal hydrides. The obtained data contribute to the understanding of phase formation processes in lanthanide beryllides and may be useful for the further development of methods for producing compacted specimens of this class of materials, for studying their properties, and for assessing their potential applicability in nuclear and fusion energy systems, as well as in other high-temperature and highly loaded systems.

#13580

Experimental campaign for deuterium extraction from PbLi by means of vanadium based permeator against vacuum in CLIPPER facility

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

In fusion reactors, tritium should be produced through neutron irradiation of a lithium-containing alloy, such as the eutectic lead–lithium (PbLi) alloy. The generated tritium must then be extracted and transferred to the Tritium Plant for reinjection into the plasma chamber. To date, this step represents a major showstopper for the realization of functional commercial fusion reactors. Consequently, experimental validation of tritium recovery technologies at laboratory scale is still a critical open challenge.

This work presents experimental studies on deuterium extraction using a permeator against vacuum (PAV), one of the most promising technological approaches. A vanadium-based prototype has already been manufactured, and its performance will be experimentally validated through a dedicated campaign at the CLIPPER facility.

The PAV consists of a rectangular PbLi flow channel enclosed by a 1 mm-thick vanadium membrane. Deuterium is dissolved in the liquid metal via a vanadium-based gas injection system and subsequently diffuses toward the membrane surface, where it permeates under vacuum conditions. The extracted gas is transported by the vacuum system to a quadrupole mass spectrometer for detection and quantification.

Experiments are being conducted at temperatures between 400 and 500 °C under low PbLi flow conditions. The results will be compared with those obtained in a previous experimental campaign using a free-surface PAV in order to assess and quantify the extraction capability of the present configuration.

#13585

Tritium breeding capabilities & experimental results at the tritium breeding research facilities by the University of Bristol & Astral Systems

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The University of Bristol and Astral Systems have established a unique tritium breeding research facility in the north of Bristol that allows the seamless testing of tritium breeder materials on a variety of different conditions under neutron irradiation.
This facility leverages the expertise of the University of Bristol on material manufacture and instrumentation design and Astral's compact neutron sources.
The facility consists of a gas management system (GMS), a breeder blanket module (BBM) and DD and DT neutron sources. The GMS has been developed to manage the purging gas flow through the breeder material during neutron irradiation. Flow, pressure, breeder temperature and gas composition can be modified in in real-time within the breeder module. Qualitative chemical analysis of the purging gas is done in real time as well as post irradiation liquid scintillation counting to quantify the tritium produced. The BBM encloses the vessel where the breeder material is encapsulated and conditions the neutrons to maximise breeding efficiency. The current neutron sources leverage Inertial Electrostatic Confinement Fusion (IECF) technology and currently is capable of generating I $\approx 1\times10^{8}$ n/s DD neutrons (E = 2.45 MeV) and I $\approx 5 \times 109$ n/s DT neutrons (E = 14 MeV).
An overview of the results obtained to date on the testing of different breeder materials will be presented as well as and status update and future steps of the whole facility.

Modelling activities using SIMMER-III code to support the LIFUS5/Mod3 Series F experimental campaign

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

This work presents the simulation models implemented for the analysis of LIFUS5/Mod3 Series F experimental campaign. The LIFUS5/Mod3 facility, located at the Brasimone ENEA Research Centre, is a Separate Effect Test (SET) facility specifically designed to provide experimental data on the chemical interaction between Lithium-Lead and water under Water-Cooled Lithium-Lead (WCLL) Breeding Blanket accident conditions, with reference to the in-Box-LOCA scenario. These transients are characterised by strong thermal-hydraulic phenomena coupled with rapid chemical reaction kinetics, requiring advanced multi-phase, multi-component computational tools for accurate predictions. In this framework, the SIMMER-III and SIMMER-IV codes were selected as reference tools for simulating the PbLi - water interaction. Experimental campaigns at LIFUS5/Mod3 have played a crucial role in validating these codes, both for chemical reaction modelling and overall thermal-hydraulic behaviour. Within the FP9 programme (2021–2024), the facility has been employed for the Series F campaign, which includes ten tests aimed at extending the experimental database and assessing the predictive capabilities of the codes. The modelling approach adopted for Series E was updated and refined, improving nodalization, boundary conditions, and reaction modelling, and then applied to the Series F tests. A comprehensive experimental analysis highlights the main differences observed compared to previous campaigns. Special emphasis is placed on SIMMER-III validation methodology, which integrates qualitative engineering judgment with quantitative evaluation through selected figures of merit, supporting a structured assessment of code accuracy for WCLL in-Box-LOCA scenarios.

Upgrading heat transfer correlations for plasma-facing components with one-side heated rectangular cooling channel.

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The Nukiyama curve describes the different regimes of the convective heat transfers for a heated tube with a coolant circulated inside. The correlations used to construct this curve have been largely studied in the past and updated through the years for tube with circular section and especially for the case of anisotropic heating of the tube [1]. Nevertheless, innovative concepts for plasma facing components (PFC) in fusion device are increasingly turning to cooling channel with rectangular section. Up to now, neither dedicated experiments have confirmed the applicability of the existing correlations to this geometry nor specific correlations have been done to derived the heat transfer in the case of a rectangular cooling channel. Usage of relevant correlations is of prime importance to define proper boundary conditions and provide relevant thermomechanical calculations for PFC dimensioning.
The present work aims to fill the gap and propose an update of the existing empirical correlations by adjusting the coefficients based on experimental results for plasma-facing components with one-side heated rectangular cooling channels. Dedicated experimental campaigns were thus performed on HADES Electron Beam high heat flux facility (CEA-Cadarache) for both copper (CuCrZr alloy) and stainless-steel (316L) mock-ups with rectangular cooling channel, investigating various geometric and hydraulic parameters (velocity, inlet temperature and pressure). The temperature data from embedded thermocouples close to the cooling channel have been systematically compared to finite element method (FEM) calculations based on both existing and optimized correlations to describes the heat transfer.
The update of the correlations focuses on the Sieder-Tate, Thom-CEA and Tong (75)-CEA correlations [1], which appears as the most impactive ones for the considered cases. The correlations have been optimized mainly based on stochastic gradient method, searching to minimize the discrepancy between the experimental temperatures from thermocouples and the simulated ones out of FEM calculations. Based on this study, new set of coefficients have been thus re-assessed which improves the matching rate with the experimental data regarding the case with classic correlations (for circular tube). This upgrade allows a better prediction of the convective heat transfer and critical heat flux threshold for plasma-facing components with one-side heated rectangular cooling channels at relevant thermal-hydraulic conditions with regard to water cooling (i.e. for a subcooled flow boiling regime).

[1] J. Boscary et al, Critical heat flux of water subcooled flow in one side heated swirl tubes, International journal of heat and mass transfer 42, pp 287-301, 1999.

Analysis of Residual-Magnetization-Induced Electromagnetic Loads in the DEMO Breeding Blanket System

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The DEMO breeding blanket system is largely based on EUROFER97 steel, a soft ferromagnetic material that may retain remanent magnetization after plasma operation. During maintenance and disassembly phases, when all coils and plasma are switched off, this residual magnetization can still generate electromagnetic (EM) forces, potentially affecting blanket handling and removal operations. This work presents an assessment of EM loads induced by remanent magnetization for the DEMO breeding blanket segments under representative maintenance scenarios.

A full 360° finite-element electromagnetic model of the blanket system was developed using ANSYS EMAG. Blanket modules were modeled with a homogenized material approach accounting for the effective EUROFER97 content, while toroidal air gaps between adjacent segments were explicitly included to capture magnetic reluctance effects. The homogenization strategy was verified against a heterogeneous material representation, showing very good agreement in the predicted forces and moments.

Several non-axisymmetric configurations were analyzed, including the removal of the Central Outboard Blanket (COB) and the progressive removal of multiple blanket modules within a single toroidal sector. In addition, a simplified 360° model was employed to estimate the electromagnetic “recall” force acting on the COB during its vertical extraction through the upper port, combining numerical results with analytical extrapolation along the full removal path.

The results show that EM forces and moments due to residual magnetization are typically of the order of 10^2 N and 10^3 Nm, several orders of magnitude lower than operational EM loads and structural design limits. The recall force during COB removal is negligible compared to blanket weight and remote-handling capabilities. These findings indicate that residual-magnetization-induced EM effects do not represent a critical issue for DEMO blanket maintenance and disassembly.

#13657

Physics, technology, and economic considerations for DD startup of DT fusion power plants

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

A deuterium-deuterium (DD) startup is a potential strategy to start a deuterium-tritium (DT) fusion power plant (FPP) without relying on an external tritium supply (startup inventory). The feasibility of this approach depends on a range of physical and technological requirements that have not been comprehensively addressed yet. In this work, we present an analysis applicable to any generic magnetic confinement fusion device. The design space is explored to identify the
operational regime under which a DD startup could successfully start a DT FPP while also satisfying economic and physics constraints. The analysis integrates the main physical, technological, and economic drivers, including plasma
density, temperature and confinement times, tritium breeding with DD and DT neutron spectra, tritium process time and fuel cycle technologies, tritium price, and economic and financial metrics such as the overnight capital cost, levelized
cost of electricity, and net present value. Among the strategies considered, the most favorable pathway is a gradual transition from DD to DT via a tritium-seeded startup, in which bred tritium is re-injected as it becomes available. This
approach is compatible with standard DT fuel-cycle architectures and can generate sufficient in situ tritium to reach 50:50 DT operation on a timescale of less than one year, with only marginal economic deviation from an immediate
50:50 DT startup. A key challenge, however, is the large auxiliary power required to sustain an initially 100% D plasma (~100–200 MW). Seeding the initial fuel mixture with a modest tritium fraction (10 - 20% T) reduces the auxiliary
power required and accelerates the transition to full DT operations. Thus, if the auxiliary power requirements limit the feasibility of a pure DD startup in a given design, a tritium-lean, tritium-seeded strategy could substantially reduce the
external startup inventory to ~ 100 g while reaching 50:50 DT operation in less than 100 full power days.

Molten Fluoride Salts for Breeding Blankets: Tritium Transport Gaps and Corrosion Modeling

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Molten fluoride salts are potential candidates for tritium breeding blankets and cooling systems in future nuclear fusion reactors due to their high boiling point, favorable heat transfer properties, and neutronic performance, making them highly effective for combined breeding and cooling functions. Employing molten salts, however, relies on solving complex challenges regarding tritium management and material compatibility. The two issues are coupled: since hydrogen isotopes in fluoride melts can exist as HF/TF as well as H2/T2, their transport shifts the salt redox condition, therefore they can modulate the thermodynamic driving force for structural material dissolution in contact with the salt. This work presents an integrated analysis of tritium transport in molten salts and corrosion. FLiBe is the leading fluoride salt option in fusion reactor concepts due to efficient tritium breeding, while FLiNaK is commonly used as a surrogate for FLiBe in corrosion studies, due to similar thermophysical behavior and easier handling.
A critical review has been carried out to identify fundamental gaps in tritium transport properties in FLiBe. Tritium diffusivity data in FLiBe is slightly scattered, yet remaining fairly bounded in the range of interest for FLiBe systems (800 – 900 K). As regards solubility, less data is available: experiments present highly scattered values, and the chemical form of hydrogen has a strong impact on solubility. A few experiments were run with tritium, as hydrogen or deuterium were used as proxy. Therefore, isotopic effects on diffusivity and solubility are discussed as well. To tackle the issue of material compatibility, a numerical model has been developed to investigate the corrosion of structural materials in contact with molten fluorides. The model simulates mass transfer and chemical and electrochemical kinetics at the metal and salt bulk and at salt-metal interface in a simplified 1D geometry. This model serves as a tool for parametric sensitivity studies. It allows for the identification of dominant corrosion regimes (initial fast attack vs. long-term slow diffusion limited by solid species migration) and phenomenological regimes. Overall, this work sets the basis for integrated studies on molten salts in breeding blankets.

#13667

Experimental study of PbLi MHD flow in a circular pipe

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The flow of lead-lithium in liquid metal breeder blankets is governed by magnetohydrodynamic (MHD) effects. As the electrically conductive fluid moves through the plasma-confining magnetic field, electrical currents are induced within the fluid. These currents interact with the magnetic field, generating strong Lorentz forces, which result in peculiar flow patterns and lead to pressure drops several orders of magnitude larger than those observed under hydrodynamic conditions.
In the case of the Water-Cooled Lead-Lithium (WCLL) blanket concept, nuclear heat is removed by cooling pipes immersed in the breeder zone. This design means that the circulation of the liquid metal is only needed for tritium extraction. As a result, the forced flow of PbLi in the breeder units is reduced to minimize pressure drops
To support the development of the WCLL blanket, a deep understanding of the underlying physics of PbLi flows occurring in the blanket and numerical predictive tools validated against experimental data are required. To that extent, the MaPLE (Magnetohydrodynamic PbLi Experiment) facility was recently recommissioned at KIT.
In the present study, the first experiments conducted in MaPLE at KIT are presented. For testing electrical potential and pressure drop measurements at high temperatures and under strong transverse magnetic fields, we consider the well-known and fundamental MHD problem of flow in a circular pipe of constant cross-section with conducting walls. Experiments were conducted for MHD flows in a uniform magnetic field and for flow entering and exiting the magnet in a fringing field. Besides validating diagnostic tools for future experiments, the data collected complements previous experimental work and expands the database for benchmarking numerical codes with results obtained for another wall conductance ratio and distribution of the magnetic field along the axis of the pipe that differs from previous data.

#13705

Preliminary neutronics analysis of Tribreed blanket mock-up for advanced ceramic breeders

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Experimental validations of tritium production with spectral neutron flux are needed to guarantee tritium fuel self-sufficiency. As part of UKAEA's Lithium Breeding Tritium Innovation (LIBRTI) program, a prototype tritium breeding mock-up is developed to validate tritium production models in Tritium Breeder (TriBreed) project. This study presents preliminary neutronic analysis conducted using particle transport codes to evaluate breeding performances of advanced breeding materials in the mock-up. The objective of this simulation study is to calculate the neutron flux distribution across the experimental assembly and evaluate the radio-activation of structural components. The mock-up had a HDPE (high density polyethylene) structure contained a representative breeder zone, namely steel, Pb (neutron multiplier), and ceramic breeders. A single-crystal diamond detector with 6LiF film converter and activation foils were also contained in the mock-up for tritium and neutron measurements. Tritium production rates in lithium oxide (Li2O), advanced octa-lithium ceramic breeders, and Li2CO3 diagnostic pellets were calculated to design experiments, including high-flux neutron irradiation and subsequent tritium analyses utilizing an accelerator mass spectrometer (AMS) and liquid scintillation counter (LSC). As the mock-up is planned to be irradiated in High Flux Accelerator-Driven Neutron Facility at Birmingham University using reaction between accelerated proton and a solid metal lithium target, spectrum neutrons were generated with a neutron production rate of 10^12 n/s in the calculations. Neutrons transport was calculated by using a combination of the PHITS code and OpenMC. After neutron irradiation for 24 hours, tritium production per mass were calculated to be >1000 fg/g-breeder, which are substantially above the detection limits of the AMS and LSC, enaling accurate measurements of tritium production. Activation foils and diagnostic pellets are placed on stringers so that the irradiated materials are rapidly taken out for following measurements without disassembling the whole mock-up. Activation analysis using D-chain code was performed to evaluate the radio-activation of individual components and to determine appropriate cooling times. By measuring tritium production rates and evaluating the calculation to experimental (C/E) ratios, this study aims to make a prototyping device to validate models for tritium production and release properties in novel ceramic breeder materials, ultimately supporting the commercial viability of future fusion reactor designs.

#13737

CFEDR Tritium Fuel Cycle Design and Optimization

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The China Fusion Engineering DEMO Reactor (CFEDR) constitutes a pivotal link in China's three-stage fusion energy roadmap, bridging the gap between experimental fusion devices and future commercial reactors. A prerequisite for the long-term operation of CFEDR is the realization of tritium self-sufficiency, which remains one of the most critical challenges in fusion energy systems.

In this study, a fully integrated tritium fuel cycle architecture is developed for CFEDR, incorporating the internal fuel cycle, the external fuel cycle, and the tritium safety and confinement system. Using the mean residence time (MRT) approach, tritium transport and inventory distribution across all subsystems are first quantified under nominal operating conditions. The effects of key operational parameters—including the plasma burn fraction, plant availability, and fueling efficiency—are then systematically investigated through sensitivity analyses, with particular focus on their influence on the required initial tritium inventory, the time to reach tritium self-sufficiency, and the tritium doubling time. Building on the reference fuel cycle model, a 100% Direct Internal Recycling (DIR) operational strategy is proposed and evaluated. A deuterium–tritium mixed fuel cycle model is employed to demonstrate the technical feasibility of this approach. The results indicate that full internal recycling can substantially reduce the initial tritium inventory while also simplifying the fuel cycle configuration and lowering both capital and operational costs. To address the limitations and uncertainties inherent in system-level simplified models for highly complex tritium processing schemes, this work further advances a multi-degree-of-freedom, high-fidelity modeling framework. Key subsystems with large tritium throughput and complex transport phenomena are treated using detailed local models coupled to the overall system analysis. As a representative example, a strongly coupled flow and mass-transfer model has been developed for the palladium-membrane permeator in the tritium extraction and purification (TEP) system, enabling high-accuracy simulations of hydrogen isotope separation from impurities. Experimental validation confirms the robustness, accuracy, and engineering applicability of the proposed model.

Overall, this work provides an integrated and quantitative framework for tritium self-sufficiency assessment in CFEDR and offers valuable theoretical and engineering guidance for the design and optimization of tritium fuel cycle systems.

Interpretable Surrogate Modeling and Cost Optimization of Wet Scrubbers' Detritiation System in Nuclear Fusion Fuel Cycle

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

ITER is a large-scale tokamak project designed to demonstrate the scientific and technological feasibility of fusion fuel. Because tritium is used as a fusion fuel and can be released from process equipment, ITER requires robust Detritiation Systems (DS) to minimize atmospheric emissions for plant operation. DS located at the end of the fusion fuel cycle, the DS typically integrates a catalytic recombiner that oxidizes tritium to tritiated water with a wet scrubber column that captures the resulting tritiated water vapor prior to discharge. Rigorous wet scrubber models designed with Aspen Plus flowsheets can represent this highly nonlinear gas–liquid isotope exchange process, but repeated techno-economic assessment (TEA) and optimization become computationally complex when embedded in plant-wide studies.

This study develops interpretable surrogate models for the wet scrubber that approximate detritiation performance and annual total cost and can be embedded in optimization workflows. A proprietary simulation-based framework generates a design-of-experiments dataset via advanced sampling techniques over multi-dimensional operating parameters. For each sampled condition, internal sizing calculations are performed to determine stage num ber, packing height, column diameter, and pressure drop using standard hydraulic and mass-transfer correlations, enabling consistent cost evaluation. The resulting dataset contains two outputs relevant to operation and assessment: detritiation factor and annual total cost. Diverse surrogate modeling architectures were evaluated for accuracy, optimization compatibility, and interpretability. Evaluation was conducted via performance metrics such as R2 and RMSE.

The resulting surrogates enable rapid screening of wet scrubber designs and cost-optimal operation under detritiation constraints without repeatedly solving full Aspen simulations, thereby reducing computational time and cost in iterative TEA. This surrogate model can support upper-level system cost minimization by accelerating flowsheet evaluations, enabling efficient exploration of design-operation tradeoffs, and providing interpretable relationships between operating conditions, detritiation performance, and annual cost.

#13758

TRICYS for Reproducible Tritium Fuel-Cycle Simulation: A CFEDR Case

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Tritium fuel cycle (TFC) simulations often combine multiple domain-specific models. Because many existing studies connect these models using ad hoc scripts, interfaces tend to drift over time, underlying assumptions become obscured, and parameter sweeps are difficult to reproduce. To formalize this process, we developed TRICYS (TRitium Integrated CYcle Simulation), an open-source framework for reproducible TFC modeling. Rather than relying on custom glue code, TRICYS defines execution scenarios and tool configurations entirely through declarative JSON/YAML files. The platform supports multi-fidelity analysis by coupling a fast 0D system-dynamics solver—used for plant-wide tritium inventory tracking—with external high-fidelity component models. It currently integrates Aspen Plus for process modeling and Modelica for dynamic simulation, maintaining a modular architecture to accommodate additional solvers. During execution, TRICYS automatically logs all input and output provenance data and archives the generated artifacts, guaranteeing that any workflow run can be strictly inspected or rerun. To maintain physical consistency across coupled tools, the framework features built-in regression testing that evaluates plant-level mass conservation and inventory closure, immediately flagging any anomalies introduced by model updates. We demonstrate TRICYS using a representative CFEDR (China Fusion Engineering Test Reactor) fuel-cycle scenario. This case study confirms that TRICYS effectively executes complex, multi-tool TFC workflows while providing the rigorous traceability required for extensive parameter scanning and uncertainty quantification.

Hydrogen Isotope Transport in EUROFER: Experimental Testing and Numerical Modeling of Anti?Permeation Coatings

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The minimization of tritium losses through structural materials is a key challenge for nuclear fusion reactor technology. In the European DEMO framework, EUROFER ferritic-martensitic steel is the reference structural material and is significantly permeable to hydrogen isotopes. To reduce tritium absorbed and permeated through the material, the adoption of anti-permeation barriers deposited on the steel is foreseen.

In this work, a comprehensive study of deuterium permeation through different EUROFER samples, comparing the performance of bare material with samples coated using different anti-permeation barriers, is presented. Experimental permeation tests were carried out exploiting PERI-II, a gas-driven permeation device located at ENEA C.R. Brasimone, in the temperature range 300-550°C and pressures 0.1-1000 Pa to quantify the steady-state flux and derive the permeation properties and associated permeation reduction factors (PRFs) for each coating.

The results show differences in permeation behavior depending on the applied barrier, with coatings demonstrating very significant suppression of deuterium permeation. These findings are complemented by dedicated numerical simulations, which model hydrogen isotopes transport including trapping within the sample, providing insight into the underlying transport kinetics.

Equivalent MHD boundary condition for computationally efficient numerical simulations of liquid metal systems with conductive walls for nuclear fusion

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Liquid metal (LM) systems for magnetically-confined nuclear fusion applications will operate under intense magnetic fields of the order of several T. These induce electric currents that result in Lorentz forces generating, e.g., for closed-channel flows, altered velocity profiles and large pressure losses. This is especially true when the channel walls are electrically conductive. A measure of the impact of the magnetic field is the the Hartmann number Ha = B_0Lsqrt(pσ/μ), which can be higher than Ha > 10^4 in fusion reactors. The formation of the so-called Hartmann Boundary Layers (BL) is inherently connected with these effects. The thickness of such BL scale inversely with Ha.

The appropriate resolution of the Hartmann BL has long been recognized to be essential in LM MHD numerical simulations to properly capture these flow features and the associated phenomena that affect the component performance.

However, it requires computationally expensive models due to two main reasons. First, the need for fine meshes that can resolve the Hartmann BL. Second, the need to account for the electromagnetic coupling between the fluid and the enclosing solid structure. The resulting computational cost results prohibitive.

We present the implementation of an equivalent MHD boundary condition (BC) for the electric potential. The BC models walls of finite electrical conductivity and thickness via a thin-wall approximation and optionally incorporates the
effect of unresolved Hartmann BL through an equivalent additional sheet conductance (Leboucher et al. 1999). The objective is to reduce computational cost while preserving accuracy for velocity distributions and MHD pressure losses. For comparison, the BC is developed and tested both in the open-source OpenFOAM software and in COMSOL multiphysics ®.

Validation on canonical MHD duct configurations with finite wall conductivity reproduces velocity profiles and pressure losses across a range of Ha numbers achieving, e.g., a reduction of 95% of the computational time. Also, the
application to cases with developing flow in fringing magnetic field showcases for the first time a favorable trade-off between accuracy and computational cost. Single-region simulations are therefore enabled as for suitable design
studies. Future work includes application to complex fusion blanket geometries and free surface flows for plasma-facing components.

Design rationale and functional architecture of the Water Lead Ceramic Breeder (WLCB) Breeding Blanket for EU-DEMO

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Developing a reliable Breeding Blanket (BB) is a primary requirement for the EU DEMO fusion power plant. The system must ensure tritium self-sufficiency, efficient heat removal, and structural integrity under extreme thermo-mechanical and electromagnetic loads, while complying with rigorous safety constraints. Within the European DEMO program, the Water Lead Ceramic Breeder (WLCB) BB concept combines the technological maturity of water cooling with the high neutronic performance of ceramic breeders and lead-based neutron multipliers. This design exploits the advantages of water-cooled systems while securing adequate tritium breeding through advanced ceramic pebble beds.

The WLCB utilizes pressurized water at Pressurized Water Reactor (PWR) conditions to extract heat from the blanket and transfer it to the power conversion system. Its architecture follows a Single Module System (SMS) approach, enhancing the structure's ability to withstand electromagnetic loads while maintaining neutronic efficiency. The First Wall (FW) features a multilayer structure of tungsten plasma-facing armor and a Eurofer wall with embedded cooling channels. To ensure robustness against Loss of Coolant Accidents (LOCA), the FW casing is reinforced and connected to a Back Supporting Structure (BSS) hosting the coolant and tritium carrier manifolds.

During conceptual development, several breeding zone architectures were investigated, ranging from breeder-in-tube to heterogeneous assemblies. The current reference design houses the ceramic breeder and lead multiplier in dedicated cladding boxes. This modular solution improves manufacturability and ensures the required packing factors for functional materials. These boxes feature integrated fins that maintain a controlled gap between the boxes and the cooling plates. Crucially, these gaps serve as a communication volume in the event of an in-box LOCA. This environment is specifically designed to be connected to the Tritium Extraction and Recovery (TER) system, which could be equipped with safety accessories to mitigate peak pressure spikes and their consequences. Thermal power is removed via cooling plates linked to an integrated manifold, while tritium is extracted from the ceramic breeder using a helium purge gas. The WLCB design rationale balances neutronic performance, thermal-hydraulic efficiency, and safety.

This paper presents the design objectives and architectural evolution of the WLCB concept, discusses remaining open gaps, and outlines the design development strategy and R&D activities supporting the current reference configuration.

#13789

In situ measurement of Tritium permeation behavior in High temperature liquid LiPb

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Eutectic lithium–lead (LiPb) alloy is considered one of the most promising candidate materials for liquid breeder blankets in fusion reactors because it can simultaneously serve as a tritium breeder and a coolant. For LiPb blanket systems, accurate knowledge of the transport properties of hydrogen isotopes is essential for tritium transport analysis and the design of tritium recovery systems. However, the available experimental data on these key parameters remain limited. In this study, a newly developed apparatus based on the permeation pot method was employed to investigate the transport behavior of hydrogen isotopes in liquid LiPb over the temperature range of 250–600 °C. Permeation experiments were carried out using protium and deuterium as surrogate isotopes, and Li17Pb83 with natural lithium isotopic abundance, fabricated in China, was selected as the test material. Hydrogen was also employed to control the oxygen activity in liquid LiPb. The experimental results demonstrate that hydrogen isotopes migrate through liquid LiPb in atomic form and exhibit transport characteristics of low solubility and high diffusivity. On this basis, the solubility and diffusivity coefficients of hydrogen isotopes in liquid Li17Pb83 were determined. The influence of impurities on tritium speciation, particularly under different oxygen conditions, was taken into account in the analysis. These results provide important experimental support for tritium transport modelling and the engineering design of LiPb blanket systems.

Tritium inventory estimate of the EUROfusion neutral-beam-driven Volumetric Neutron Source and its dependence on the hydrogen isotope allocation to torus and neutral beams

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The Volumetric Neutron Source (VNS) studied by EUROfusion is a medium-sized tokamak in which continuous neutral beam injection (NBI) drives deuterium–tritium (D–T) fusion and the plasma current. In order to maximise the fusion rate for a given NBI power it seems beneficial if NBI and the torus plasma use different hydrogen isotopes, either D for NBI and T as the torus feed gas, or vice versa. The tritium inventory in the whole fuel cycle depends on this choice and its estimate therefore provides critical information for an informed decision which isotope to use where in the system. This contribution describes the methods and assumptions used for the estimate and it gives the predicted inventories obtained for a number of reference cases; these cases are all variants of D NBI beams fired into a T plasma, T beams fired into a D plasma, and D/T beams fired into a D/T plasma. The calculated tritium inventory outside the tritium storage stays below 1 kg for all investigated cases and varies by less than a factor of two between the isotope assignment options due to competing effects. The isotope separation system and the cryopumps of the torus and the NBI boxes constitute the largest tritium inventories.

Pre-test analysis of experimental In-Box LOCA in LIFUS5/Mod4 facility using the SIMMER/RELAP coupling tool

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Among the critical aspects of the Water-Cooled Lithium Lead Breeding Blanket design, the iIn-Box Loss of Coolant Accident (LOCA) stands out as one of the most significant ones. Over the years, considerable scientific effort has been devoted to understanding the dynamics of Lithium-Lead and Water interaction, as well as the broader consequences this could impose on the plant system. This research has been pursued through both experimental and computational approaches. Given the complexity of simulating two working fluids across multiple phases, and the necessity of capturing the chemical reactions between Lithium-Lead and Water, the SIMMER code was selected as the most appropriate tool, owing to its original development for the analysis of accidental scenarios involving Liquid Metals. However, the SIMMER code is not meant to perform a system-level numerical assessment of the accident at full-plant scale. To address this limitation, a coupled methodology between SIMMER and RELAP has been established, exploiting RELAP5 code capabilities to model the full extent of the plant affected by the accident. From the experimental point of view, with the aim of broadening the understanding of the Water and Lithium-Lead interaction through an integral test facility, the LIFUS5/Mod4 facility has been designed and is planned to be constructed at the ENEA Brasimone Research Centre. Designed on a 1:1 scale with the ITER Lithium-Lead Loop, this facility is going to be fully representative of the thermal-hydraulic phenomena expected to occur during the accident. In this framework, the work presented here represents an application of the coupling methodology to the pre-test analysis of the LIFUS5/Mod4 facility. The updated nodalization of LIFUS5/Mod4 developed for the RELAP5 code and the new nodalization of the test section developed for the SIMMER code have been coupled to perform the analysis of a reference experimental test. This numerical investigation provides valuable insight for the planning of the future experimental campaign, illustrating the effects of the interaction between the two fluids within an integral test facility. The results obtained are expected to be of significant importance in supporting the design and execution of the experimental campaign.

Effect of tilt angle on the liquid metal MHD mixed convection in a separately cooled breeding blanket with helical cooling pipes

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Separately cooled liquid metal (LM) breeding blanket (BB) are a class of components currently under development for near-term deployment in technological demonstrators and fusion pilot plants. The Water-Cooled Lead Lithium (WCLL) is the candidate BB concept pursued by the European fusion programme and relies on the eutectic lead lithium alloy as breeder, pressurized water as coolant, and Eurofer as structural material. This combination of design choices results in a slow-paced LM flow (4000°C/m). Previous studies focused on equatorial BU have revealed the emergence of a magnetohydrodynamic (MHD) mixed convection flow composed by two distinct regimes: both affected by buoyancy despite the magnetic dampening. The objective of this study is to explore the role of buoyancy in the MHD flow developing in the BUs which exhibits a significant inclination (non-null tilt angle) with respect to the equatorial plane.

For this purpose, we use the computational fluid dynamics code ANSYS CFX. We analyze the WCLL BUs in the outboard blanket close to the interface with the inboard (which is tilted downward) and with the divertor (tilted upward) as limiting cases. Simulations are performed at reduced magnetic field intensity while preserving the field topology and the overall force balance to predict the flow features appearing in operative conditions. The water coolant within the BU helical pipes and FW square channels is modelled through convective boundary conditions. Both surface and volumetric heat load are considered to predict the thermal-hydraulic performance of the system.

Buoyancy is found to play a significant role affecting the velocity and pressure distribution in the BU; nevertheless, the fundamental characteristics of the flow regime are confirmed. The BU tilted upward exhibits a moderate degradation in the heat transfer performance, but the maximum temperature of the structural material (522°C) remains comfortably within the upper limit of the temperature window (550°C).

System-level modelling for the EU DEMO Water Lead Ceramic Breeder Breeding Blanket

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

One of the Breeding Blanket (BB) concepts proposed for the EU DEMO tokamak is the Water Lead Ceramic Breeder (WLCB), which brings together the high efficiency of water cooling with the excellent breeding properties of ceramic breeders. Its design, and in particular the verification of safety requirements, is based on the predictions of numerical codes, both at the local and system levels. Such codes require careful verification (e.g. code-to-code benchmark) and validation to guarantee the reliability of their predictions.

In the present work two system-level models are developed for the WCLB water cooling system, one based on the well-established RELAP5 nuclear code, and one based on the more recent GETTHEM code developed at Politecnico di Torino and already employed to support the design of other BB concepts. A benchmark of the two models is carried out under nominal operating conditions.

Having verified the newly-developed models, they are then applied to accidental analysis, namely an in-box LOCA from one cooling plate, causing the release of water inside the BB segment. The models are extended to include the purge gas circuit, which may be used as venting circuit to connect the in-box volume with the pressure suppression pool, effectively mitigating the overpressurisation and avoiding overcoming the maximum allowable pressure in the BB box. The models are then used to perform sensitivity studies on different system parameters.

#13845

Current Canadian perspective on tritium R&D aspects of fusion energy deployment

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

With the broader wave of fusion energy deployment efforts, whether by commercial fusion companies, national programs or international organizations, the significance of tritium research and development work has reached an all-time high. The fusion reaction of deuterium and tritium isotopes remains the overwhelmingly chosen approach for reaching fusion energy commercialization at industrial scale. Canada, with its network of national labs, well established and emerging fusion companies, and utilities with capabilities to handle kilogram scale tritium, offer a unique and fertile ecosystem for research.

Recently, flagship developments have added new perspectives. These include the ongoing design and building of UNITY-2, a prototypical D-T fusion fuel cycle R&D platform; the desire of Ontario Power Generation' planning efforts to scale-up tritium handling and shipments to meet the growing global demand for tritium; the ongoing programmatic efforts of the newly announced Canadian Centre for Fusion Energy; and the strong international attendance and engagement demonstrated at the 14th International Conference on Tritium Science and Technology in Ottawa, Canada.

The combination of the global fusion momentum and the new Canadian developments have created a fertile ground in many areas related to tritium research ranging from shipping and transportation, storage and clean-up, isotopic separation, detritiation and recycling and interaction with materials.

This talk will provide a holistic overview of Canada's tritium R&D activities, highlighting both the challenges and opportunities that support broader national R&D efforts. It will also touch on how Canada is strengthening its role on the global stage through active collaboration with an international network of partners

#13860

Optimization of the manifold design for the WCLL TBM via MHD analysis

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

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 significantly the flow distribution and creates increased pressure drop compared to hydrodynamic conditions. Hence, the necessity of including MHD phenomena in the analysis of liquid metal flows for applications in fusion technology.
Experimental and numerical analyses of 3D MHD flows in a WCLL test blanket module (TBM) showed that the manifolds are key and possibly critical components for the performance of blanket modules, since they create the major contribution to pressure drop and their geometry is decisive for determining PbLi flow partitioning among BUs.
Recent numerical simulations and experimental data obtained using a scaled WCLL TBM mock-up, designed according to the original design, found a remarkably non-uniform distribution of the liquid metal in the BUs along the poloidal direction, where the first and the last one receive most of the flow, while the purge flow in the central BUs is almost negligible. Such flow partitioning is highly unfavorable for efficient removal of tritium generated in the blanket by the interaction of fusion neutrons with the PbLi. The reason of this unsuitable flow partitioning has been identified as the specific geometry of the manifolds. In the original design, the baffle wall that separates feeding and draining ducts remains at the same toroidal position along the poloidal direction, although the flow rate in the manifold channels is changing. The shortcomings of the original design motivated analytical and numerical studies with the aim of optimizing the manifold geometry by finding the best positions of the baffle walls. According to the outcomes, a new CAD design has been developed, which serves as basis for the present numerical analysis of electrically coupled MHD flow in a TBM mock-up. The obtained results confirm that with the optimized manifold design, the flow is evenly distributed among all BUs and the MHD pressure drop is further reduced compared to the initial geometry.

Quality control of Pb-16Li alloy pre- and post- corrosion tests of materials by means of ICP-OES and DSC techniques

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Nuclear fusion systems currently under development all rely on deuterium-tritium reactions as their primary fuel for energy production. While deuterium is easily available, tritium has an extremely limited natural abundance, making its in-situ generation a fundamental requirement for a sustainable fuel cycle.

Among the main candidates as breeding blanket is the Water-Cooled Lithium-Lead (WCLL-BB), which employs molten lead-lithium eutectic (LLE, or Pb-16Li) as breeding fluid. This alloy is a eutectic mixture of lead, a neutron multiplier with excellent heat-transfer capabilities, and lithium (≈16 at%), which undergoes nuclear reactions to produce tritium.

LLE, although possessing synergistic breeding properties, poses significant challenges for the structural materials of the WCLL-BB, as prolonged exposure can promote corrosion in walls and pipes, potentially leading to safety issues such as plugging of the coolant channels by cold-spot crystallization of corrosion products. For this reason, checking the initial quality of the alloy and monitoring its chemical composition during both experimental and blanket testing is essential to ensure safe operation.

To achieve this, a combination of Inductively Coupled Plasma - Optical Emission Spectroscopy (ICP-OES) and Differential Scanning Calorimetry (DSC) is proposed, allowing Pb-16Li to be characterized before and after experimental tests from both compositional and thermal perspectives.

Coupling these analytical techniques has proven extremely valuable because of their complementary nature. ICP OES potentially provides quantitative determination of lithium content and detects corrosion products originating from a variety of materials in contact with molten LLE, such as EUROFER-97 and other ferritic/martensitic steels, both with and without coatings. However, it lacks information on the oxidation state of lithium, which is highly reactive and could oxidize to compounds such as Li_2O or LiOH during experimental tests, potentially altering the eutectic composition. By contrast, DSC complements this limitation by providing the thermal profile of the alloy, where shifts in the eutectic melting peak can be correlated to deviations from the initial composition.

Examples of quality control of LLE are presented in this work, using alloy specifically employed in static corrosion tests conducted at ENEA Brasimone Research Centre on alumina-coated EUROFER-97 samples. The results are then compared with the initial condition of the alloy prior to the corrosion tests and further assessed against Pb-16Li samples from an experimental loop facility.

In combination, ICP-OES and DSC provide a robust monitoring framework capable of capturing changes in the chemical composition of LLE, thereby improving the overall alloy characterization during experimental studies.

#13871

Experimental and Modeling Assessment of Pd–Ag Membrane Integration in a Semi-Industrial CECE-Based Water Detritiation System

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The management of tritium inventories in future fusion fuel cycles requires reliable and scalable water detritiation systems (WDS) capable of achieving high enrichment factors while maintaining operational flexibility. Within this framework, a semi-industrial demonstrative solution based on the Combined Electrolysis and Catalytic Exchange (CECE) process coupled with isotopic permeation is being developed under a national research program. The system, referred to as the Advanced Modular System (AMS), has a hydrogen processing capacity of up to 4 Nm^3/h and aims to produce either tritiated water or tritium-enriched hydrogen gas, addressing the tritium management requirements of future fusion reactors such as DEMO.

This work presents preliminary results from an ongoing experimental campaign and modeling assessment of integrating Pd–Ag membrane permeation into the CECE installation. The proposed configuration aims to enhance tritium enrichment in the liquid phase collected at the bottom of the Liquid Phase Catalytic Exchange (LPCE) column, which is extracted and fed back to the electrolyzer. Demineralized water with an initial tritium concentration of approximately 1000 Bq/L is used in the electrolyzer stack.

A controlled fraction of the hydrogen stream produced by the electrolyzer stack is diverted to a permeation unit comprising two Pd–Ag membranes configured to operate in series as a two-stage system. Due to the higher permeability of protium relative to tritium, the retentate becomes enriched in tritium and is reintroduced upstream of the LPCE column.

The preliminary results address: (i) the separation performance of the two-stage permeation unit at low isotopic fractions characteristic of CECE operation, as a proof-of-concept for future multi-stage cascades, and (ii) the operational compatibility and stability of permeation when operated in tandem with CECE. Tritium concentration profiles along the LPCE column, together with gas- and liquid-phase measurements at key locations, are used to evaluate process interaction.

A mathematical model extending the CECE mass transfer framework is being developed to incorporate isotope-dependent hydrogen transport through Pd–Ag membranes based on Sieverts' law. The model describes the coupled CECE–permeation dynamics and will be validated against experimental data.

The present study provides initial quantitative insight into the feasibility of integrating membrane permeation into CECE-based WDS, contributing to the design basis of scalable hybrid tritium enrichment systems for fusion fuel cycle applications.

Recent Study of the mechanical and thermal properties of plasma sintered Be12Ti

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Beryllium-based intermetallic compounds, known as beryllides, have attracted attention as neutron multiplier materials for fusion reactor breeding blankets. Compared with pure beryllium, they exhibit reduced swelling and significantly lower reactivity with water vapor at elevated temperatures. Furthermore, recent design studies for the JA DEMO reactor propose employing beryllides in block configuration rather than in pebble-bed form, the need for an extensive property database for beryllide blocks. In this study the present work examines how the sintering temperature influences the characteristics of $Be_{12}Ti$ blocks fabricated by plasma sintering. A comprehensive understanding of the plasma-sintered $Be_{12}Ti$ material and the processing device is therefore essential. This study provides an overview of the resulting material properties.$Be_{12}Ti$ blocks were produced by plasma sintering using chemically homogenized $Be_{12}Ti$ powder, with sintering temperatures set to 1050, 1100, and 1200 °C. Measurements of sintered density, Vickers hardness, and thermal conductivity were conducted. Based on densification behavior and practical workability, 1200 °C was identified as the most suitable sintering temperature. Compression tests were subsequently performed on specimens sintered at this temperature. In addition, hydrogen generation associated with watervapor reactivity was evaluated. Further details will be presented in the conference session.

Detailed specifications of transfer models for activated corrosion products in lead-lithium breeding blanket

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Fusion energy holds the promise of a sustainable future, but its implementation remains challenging. The EUROfusion DEMO project addresses this issue by combining the expertise of different European laboratories. The CEA's IRESNE institute at the Cadarache center has developed the OSCAR code (tOol for Simulating ContAmination in Reactors) for more than 50 years. IRESNE is tasked with developing a comprehensive tool, OSCAR-PbLi, to model the mechanisms related to the contamination caused by activated corrosion products (ACPs) in the lead-lithium (PbLi) breeding blanket. Key radioisotopes include 60Co, 58Co, 54Mn, 59Fe, and 51Cr. The paper will present the conceptual efforts directed towards identifying and expressing these mechanisms.

The bibliographic review identified key transfer mechanisms responsible for corrosion rates of Eurofer steels reaching 200 µm/y. The non-isothermal circuit contains two sections: the hot section where irradiation and corrosion occur, resulting in the release of corrosion products and ACPs, and the cold section, where these products precipitate, causing contamination and circuit clogging. The high corrosion rates are primarily related to the maintained corrosion and precipitation cycle, enhanced by particle formation, deposition, and erosion, due to loosely adhered precipitates, and by MagnetoHydroDynamic (MHD) effects, specific to magnetic confinement fusion reactors, that drive corrosion rates to higher levels.

Phenomenological models are proposed based on the bibliographic review and in-house expertise related to the development of the OSCAR-Na code for sodium-cooled fast reactors (SFRs) and the OSCAR-Fusion code for cooling-water systems of fusion reactors. These models include: diffusion-advection, Polley-Skyrme for corrosion/precipitation, Beal for particle deposition , Cleaver & Yates for deposit erosion, and the perfect-sink model for the cold or magnetic traps. 1D calculations are recommended for isotope diffusion in solids, in the case of preferential dissolution, leading to an evolving concentration profile.

These models will be incorporated into the OSCAR infrastructure to create the OSCAR-PbLi tool. This tool will be tested against experimental data to validate the choice of models and to calibrate paerameter values. Ultimately, OSCAR-PbLi will be transferred to industrial partners seeking to develop fusion technologies into industrial applications.

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#13940

Influence of Fuel Cycle Design Parameters on Tritium Balance in a Water-Cooled Solid Breeder Fusion DEMO Reactor

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

In order to discuss design parameters, such as tritium breeding ratio (TBR) and tritium permeation loss to cooling water, required for stable and continuous operation of a DT fusion reactor, it is necessary to establish an evaluate model for the influence of these parameters on the tritium balance of the fuel cycle. The mean residence time (MRT) method is an effective approach for evaluating the approximate tritium balance in the fuel cycle, as demonstrated in several previous studies. In this work, a fuel cycle system consisting of a lot of subsystems such as tokamak exhaust processing system (TEP), isotope separation system (ISS), bred tritium recovery system, and water detritiation system based on JA DEMO concept was assumed and influence of design parameters on tritium balance was evaluated by the MRT method.

The required initial loading tritium amount (ILTA) was evaluated to be 2.96 kg for a fusion power of 1.5 GW, a tritium burn-up fraction of 1.7 %, and a TBR of 1.05. Under steady state conditions, the majority of tritium inventory was concentrated in the ISS. By circulating tritium via a direct recycle loop (DRL) to bypass the ISS, the overall tritium inventory was significantly minimized. Specifically, increasing the DRL fraction to 80 %, the required ILTA was able to be reduced to 1.07 kg.
The influence of the chemical form of bred tritium within the blanket on the tritium balance was evaluated, assuming that gaseous tritium (HT and T2) permeates into the cooling water, whereas tritiated water vapor (HTO and T2O) does not. When the overall permeation rate of gaseous tritium was as low as 1 %, recovering tritium in gaseous form was effective. However, at a higher permeation rate of 10 %, recovery in the form of water vapor was more advantageous in terms of suppressing the increase in tritium inventory within the cooling water. The bred tritium recovered as water vapor is processed in the TEP.

The minimum TBR required to satisfy the desirable conditions for early fusion reactors - namely, a tritium recovery time of less than 1 year and a tritium doubling time of less than 3 years - was evaluated to be 1.05, assuming no DRL and a 1% tritium permeation rate into the cooling water.

#13958

Modelling Helium nucleation, bubble growth and transport in PbLi breeding blankets under magneto-convection

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Several Breeding Blanket (BB) concepts employ the lead–lithium eutectic alloy (PbLi) as a breeder to produce tritium via breeding reactions. These reactions generate helium (He) as a by-product and, due to the low solubility of He in PbLi, local supersaturation may occur, potentially leading to He nucleation and bubble formation. The resulting bubbles may grow and coalesce to macroscopic scales, potentially degrading thermal-hydraulic performance, tritium breeding and, consequently, affect component reliability and performance.

To investigate these effects, a helium nucleation model based on Classical Nucleation Theory has been implemented in OpenFOAM and extended with a bubble growth and transport model. This framework enables the analysis of He behavior under realistic BB conditions, accounting for complex geometry, buoyancy, and magnetohydrodynamic (MHD) effects.

This contribution presents the model workflow, its assessment against reference data, and its application to a prototypical BB geometry under realistic governing parameters. Simulations were performed under representative DEMO operating conditions, varying volumetric heat generation in the breeder and magnetic field intensity. Bubble density and average radius are estimated to provide a preliminary indication of the helium volume fraction in steady-state conditions.

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#13965

Mass transfer behavior of tritium released from neutron-irradiated LiAlO2 to Zr

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

To ensure the stable start-up of DT fusion reactors, a sufficient amount of tritium is secured in advance from external supply sources. One promising approach is tritium production by the nuclear reaction of neutrons with lithium in a high-temperature gas-cooled reactor (HTGR). The feasibility of this concept depends on the establishment of reliable confinement technologies under high-temperature conditions. As a tritium confinement strategy, a method has been proposed in which chemically stable LiAlO2 is encapsulated with Zr, which has excellent hydrogen storage capacity, within Al2O3 containers which has low tritium permeability. In our previous studies, tritium was observed to be mainly released as tritiated water vapor from neutron-irradiated LiAlO2. Therefore, understanding the chemical reactions and mass transfer between tritiated water vapor and Zr is essential for evaluating the confinement performance. In this study, neutron-irradiated LiAlO2 powder was heated with Zr spheres and tritium behavior was observed. Additionally, fundamental experiments investigated chemical reactions and mass transfer between Zr powder and water vapor.

The LiAlO2 powder sealed in a quartz container was irradiated by neutrons in the Kyoto University Research Reactor. The irradiated LiAlO2 powder was packed in a quartz reaction tube with or without Zr spheres, 0.5mm in diameter. The reaction tube was heated up to 900℃ at a constant ramping rate under Ar flow and the released gaseous tritium (HT and T2) and tritiated water vapor (HTO and T2O) were collected separately in water bubblers.

The total amount of tritium released from the LiAlO2 with the Zr spheres was 20% lower than that without the Zr spheres. This 20% reduction is attributed to the absorption of tritium into the Zr spheres. The fraction of tritium released in gaseous form was only 7.5% of the total release from the LiAlO2 alone, but it significantly increased to 42.6% when the LiAlO2 was loaded with the Zr spheres. These results indicate that the Zr spheres reduced the tritiated water vapor released from the LiAlO2, leading to partial tritium absorption and subsequent emission into the gas phase.

Zr powder, packed in the quartz tube, was exposed to 1%H2O/Ar flow at 300℃, 700℃, and 900℃. H2O and H2 concentrations were monitored by a dewpoint meter and a gas chromatograph, respectively. With increasing temperature, H2O concentration decreased, and produced H2 for 1-hour was highest at 700°C. This temperature dependence is attributed to the balance between the reduction reaction rate and the absorption rate.

#13969

The deuterium retention properties of advanced neutron multiplier materials after irradiation with D ions

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

In DEMO (Demonstration) fusion reactors, beryllides (intermetallic compounds of beryllium) are planned for use as neutron multipliers in the blanket system. The proposed blanket concept adopts beryllide blocks rather than beryllide pebbles because blocks provide higher thermal conductivity, lower tritium retention, and a simpler structural design. To effectively generate tritium (T) as fuel, the neutron reaction between lithium and neutron (n), and 9Be(n,2n)4He of nuclear reaction is used in beryllide. Since other nuclear reactions between Be and neutrons result in tritium production. The tritium-retaining beryllides can affect the safety and operation of a fusion reactor. Therefore, it is crucial to investigate the tritium retention properties due to the reduction in tritium retention and the expectation of tritium desorption behavior.

In this study, the deuterium retention properties of the beryllide (Be12Ti) were examined using an deuterium ion gun and thermal desorption spectroscopy, and the results were compared with those of pure beryllium. $Be_{12}Ti$ samples subjected to mechanical polishing and electropolishing were prepared. Under irradiation conditions, deuterium(D) ions were implanted with 3 keV, fluence was 1x1021 D/m2, and irradiation temperatures were from R.T. to 873 K. The bubble behaviors in Be12Ti and Be were observed using a transmission electron microscope with an ion gun under irradiation and annealing. In addition, thermal desorption spectroscopy evaluated apparent desorption activation energies from the electropolished and mechanically polished Be12Ti samples after the irradiation at high temperatures.

Simulating water-molten metal interaction in OpenFOAM: a methodology to evaluate LOCA loads on relevant structures.

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Recent advances in nuclear technology, for both fusion and fission reactors, have led to the adoption of various types of molten metals as the primary coolant. While low-pressure molten metals do not inherently represent a significant operational hazard, most designs employ water to extract and convert the energy produced by the system. This introduces considerable safety concerns, as a failure of the separating boundary between the two fluids, such as in-box LOCA or STGR events, could subject the machines to rapid pressure and temperature transients. In this work, we present the activities carried out at the ENEA Brasimone Research Centre on the development of CFD simulation techniques, based on the open-source software OpenFOAM, aimed at evaluating such fast transients. The research draws upon decades of experience in the experimental simulation of the interaction between water and various molten alloys and seeks to validate simulation methods and procedures for assessing the behaviour of fusion reactors under in-box LOCA accident conditions. To illustrate these principles, the methodology is applied to estimate the transient loads on the test section of the LIFUS5/Mod4 facility, which is designed to reproduce in-box LOCA accidents of the WCLL-TBM module of the ITER reactor. These loads are assessed by simulating the multiphase flow arising from the injection of high-pressure water into the molten alloy domain, after which the results are transferred to a structural mechanical simulation to evaluate the ability of the structure to withstand them. The obtained results are then discussed, along with the potential applicability of the developed technique to other scenarios, such as the assessment of structural loads during a typical STGR event.

Progress in the Design and Technology Maturation of the Breeding Blanket and Tritium Extraction Systems for the EU DEMO during its Conceptual Design Phase and Perspectives

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The Conceptual Design Phase (CDP) of the DEMOnstration fusion reactor has taken place from 2021 to 2025 in the frame of the EURofusion Consortium activities. During this phase, the design of the two driver Breeding Blanket (BB) candidates, namely the Water Cooled Lead Lithium (WCLL) and the Helium Cooled Pebble Bed (HCPB), including their Tritium Extraction and Removal Systems, have been maturated within the Work Package Breeding Blanket (WPBB). The issues identified at the end of the pre-CDP have been the starting point for the definition of the design development program during this phase. In particular, the issue of the low reliability of the systems has been driving the design efforts and has been demonstrated as one of the most challenging issues currently in fusion technologies. Yet, a very positive trend could be achieved in the different systems, which paves the way to give an integral solution for this long-standing issue. In order to reduce technical risks, the community has enlarged the BB design space and has been investigating parallel alternative blanket concepts, aiming at bridging the gap between the technology state-of-the-art and what is needed for a fusion power plant. This effort has led to a hybrid concept between the WCLL and HCPB, so-called the Water Cooled Lead and Ceramic Breeder (WLCB), as best near-term trade-off concept.

Together with the design efforts, the Technology Readiness Level (TRL) of the key technologies associated to the three BB and TER concepts has been advanced, going from first principles observations (TRL 1-2) for some of them to laboratory scale feasibility demonstrators (TRL 3-4) for the majority. This paper gives an overview of the progress reached in the design and technology development for the BB for the EU DEMO / First-of-a-Kind (FoaK) power plant, giving evidence for the technology maturation of the different systems and concludes with an outlook on the steps needed towards the full qualification of the BB for a FoaK power plant.

Real-Time Monitoring of MHD Liquid Metal Flows with Shallow Recurrent Decoders

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

State estimation in complex Magneto-hydrodynamic flows is a critical challenge for real-time monitoring of liquid metal blankets in Tokamak nuclear fusion reactors. Due to the multiphysics and highly non-linear nature of these phenomena, high-fidelity simulations are computationally prohibitive for real-time applications. To overcome this limitation, this work investigates a novel Data-Driven Reduced Order Model (DDROM) framework: the SHallow REcurrent Decoder (SHRED) coupled with Principal Component Analysis. Specifically, SHRED is employed to map extremely sparse temperature measurements to the full thermo-hydraulic state of the system. Furthermore, an ensemble of sensor configurations ensures robust state estimation and provides reliable uncertainty quantification.

The framework is tested on a 3D engineering scenario simulating a liquid metal (lead-lithium) flow past two cylinders subjected to an external magnetic field. This setup represents a geometrically simplified fusion water-cooling system, where the cooling of the plasma-heated metal is modeled by imposing a temperature boundary condition on the surface of the transverse tubes. Preliminary single-parameter studies using SHRED demonstrated a test reconstruction error lower than 5% for velocity, pressure and temperature fields.

The core novelty and major contribution of this work lie in the unprecedented two-parameter analysis, where both the intensity and the inclination angle of the magnetic field are varied simultaneously. This double-parametric variation induces severe, non-linear transitions in the flow dynamics, ranging from chaotic behavior at low magnetic field intensities to laminarized regimes at high intensities, characterized by the formation of asymmetric side layers for an inclination angle equal to 30°. SHRED reconstruction shows a remarkably low mean relative error, around 5% for the temperature, pressure, and velocity fields. This accuracy is maintained across both weak and strong magnetic fields (0.075 T – 0.300 T) and for inclination angles in the range [5◦, 30◦], where the angle test set reflects the dominant toroidal, rather than poloidal, component of the external magnetic field. These error values are only slightly larger than the lower error bound dictated by the low rank truncation, establishing SHRED as a reliable state estimator for complex and realistic engineering applications in

which multiple parametric dependencies are inherently coupled. Despite the extreme complexity of this parameter space, SHRED accurately infers the full-state dynamics for completely unseen parametric combinations. Moreover, these low reconstruction mean relative errors validate SHRED as an accurate, reliable, and real-time DDROM state estimation technique, allowing for online monitoring and control of real facilities.

Assessing the natural ortho/para conversion of protium in the T2ApIR experiment

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

Within the fuel cycle of a fusion power plant all six hydrogen isotopologues exist in a diverse range of states. The gaseous phase occurs in pumping and transport and the liquid phase in cryogenic distillation for isotope separation. The solid phase appears in injection pellets or targets. Successful handling of these isotopologues requires not only an extensive phase space knowledge but also robust, in-situ analytical tools to monitor their composition, homogeneity and purity. For the gas phase mass spectrometry, gas chromatography, and laser Raman (LARA) spectroscopy are well established methods, while absorption of infrared (IR) radiation is negligible in the gaseous phase state. However, at the high-density regime (liquid, solid) IR absorption spectroscopy becomes a suitable tool. Collisions, phonons and dimer formation induce the necessary transition dipole moment for absorption of IR light. With these effects being highly non-linear, precise quantitative analysis of isotopological and nuclear spin (ortho/para) composition demands both rigorous experimental calibration and a sophisticated model-based understanding of molecular interactions.

With T2ApIR (Tritium Absorption IR spectroscopy), an experimental setup was designed and assembled at Tritium Laboratory Karlsruhe (TLK) to perform the calibration of IR spectra as well as to exploit various analytical methods to investigate the phase space of hydrogen. The successfully performed inactive commissioning phase is coming to an end and first experiments with tritium are foreseen for the second half of 2026. The core component of the setup is a cryogenic sample cell capable of liquefying and solidifying all six hydrogen isotopologues. Dual-sided optical access allows IR and LARA spectroscopy, photography and polariscopy on liquid and solid hydrogen. The set of analytical tools gets supplemented by an external gas phase LARA spectrometer.

This presentation provides a comprehensive overview of the T2ApIR experiment. Specially emphasised are the recent results from measuring the time constant of natural ortho/para conversion of protium in T2ApIR. As IR spectroscopy is a method sensitive to ortho/para configurations, characterising this effect is a key factor for a successful high-accuracy calibration and subsequent reliable results.

#14019

Magnetohydrodynamic flow in a toroidal manifold for WCLL blankets

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The Water-cooled lead lithium (WCLL) breeding blanket is a promising candidate for the European DEMO nuclear fusion reactor. In this blanket design, the liquid lead lithium serves as breeder and neutron multiplier, pressurized water as coolant and EUROFER as structural material. When the electrically conducting PbLi circulates in the blanket, its interaction with the plasma-confining magnetic field leads to the occurrence of magnetohydrodynamic (MHD) effects. They are caused by induced electric currents and electromagnetic forces in the fluid that influence velocity and pressure distribution.

The current designs of the manifolds distributing the liquid metal into the breeding zones in the DEMO blanket and in the ITER test blanket module (TBM) present similarities. Apart from the higher complexity in the former one, some common typical geometric elements can be identified. The manifold features an inlet circular pipe connected to a toroidal distributing chamber through which the liquid metal enters openings in the back plate to supply PbLi via two circular pipes to poloidal channels that feed the breeder units.

The MHD flow in a toroidal expansion chamber of a DEMO blanket segment has been the subject of a previous analysis performed by applying an asymptotic method valid for strong magnetic fields and under the assumption of inviscid and inertialess core flow. Moreover, the so-called thin wall approximation was applied, where it is assumed, that currents leaking into the walls flow exclusively in wall-tangential direction. This simplification is applicable when the wall is significantly thinner than the characteristic dimension of the flow domain and the electric potential does not vary across the wall thickness. In order to verify if these assumptions are justified, the problem in the fluid and in the wall is analysed by full numerical simulations. The influence on flow features, depending on the strength of the applied magnetic field and on the flow rate, are investigated. The aim of this study is not only to complement and verify available results but also to get a better characterization of the liquid metal flow in a toroidal manifold from the MHD point of view and details about pressure losses in such a component.

#14022

Tritium breeding and fuel cycle research at MIT PSFC: recent results from the BABY/LIBRA team and future outlook

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

The private fusion industry is rapidly growing and pursuing aggressive timelines for the construction and operation of deuterium-tritium fueled pilot plants. Tritium breeding blanket and fuel cycle technology is at a low technical maturity, especially relative to the tritium throughputs fusion power plants will require. The MIT Plasma Science and Fusion Center leads an extensive effort to better understand tritium breeding in fusion conditions, especially in liquid breeders like molten fluoride salts. In this talk, we will provide an overview of the highlights of our program, and discuss important recent results, especially as they pertain to the BABY and LIBRA tritium breeding campaigns.

The MIT PSFC team has performed tritium breeding experiments in small volumes of FLiNaK, ClLiF, and FLiBe (100 mL - 1L) exposed to a DT neutron source, and recently has begun expanding efforts to also investigate tritium breeding in PbLi. We will present these results and show how experimental accounting of tritium compares to modeled predictions for tritium production, and discuss the implications of these results for scaled-up breeding blankets. We will also discuss the recently completed MIT FLiBe laboratory, which will enable tritium breeding experiments in larger (100s of L) volumes of FLiBe and ClLiF salts. Progress on those scaled-up experiments will be presented.

As part of a FIRE collaborative project led by U. Wisconsin Madison, the MIT PSFC team is also partnering with SHINE Technologies and Commonwealth Fusion Systems to run FLiBe and PbLi breeding experiments at SHINE's FLARE neutron source. This will enable tritium breeding experiments in a much more intense neutron source, getting us closer to understanding tritium breeding and tritium science in truly prototypic fusion neutron environments. Initial experiments are planned for Fall 2026 and progress updates will be provided.

Demonstrations of tritium breeding are just one part of the R&D program required to derisk fusion pilot plant fuel cycles. In this talk, we will also discuss our work on tritium permeation barrier development, fuel cycle modeling, analysis of tritium retention in

materials, corrosion of materials simultaneously exposed to molten breeders and irradiation, and fuel cycle component development. Finally, we will assess the gap between our existing research program and the derisking still needed to design and build tritium self-sufficient fusion power plants with robust breeding blanket and fuel cycle technologies.

Modelling of hydrogen mass transport in the Permeator Against Vacuum technology for the Tritium Extraction and Removal system of the EU DEMO WCLL blanket

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Achieving tritium self sufficiency is a central requirement for electricity production from future nuclear fusion reactors. The Tritium Extraction and Removal (TER) system of the EU DEMO Breeding Blanket (BB) must therefore be designed to maximize tritium extraction while keeping the inventory low, which calls for reliable modelling tools to support the design of the components involved. Within the Water Cooled Lead-Lithium (WCLL) BB concept, tritium is recovered from the liquid lithium–lead (PbLi) breeder by means of the Permeator Against Vacuum (PAV), selected as the reference technology for the extractor.

A previous work showed that a recently developed CFD derived Sherwood number correlation for fully developed PbLi flow, when implemented in the system level GETTHEM model of the PAV, significantly underestimates the tritium extraction efficiency measured during experiments on a PAV mock up installed at the TRIEX-II facility in ENEA Brasimone research centre. These results highlight the critical role of the enhanced transversal mass transport of tritium within the PbLi in the entrance region of the channels in the accurate modelling of tritium transport.

In this work, a dedicated CFD study is carried out to investigate hydrogen mass transfer in the developing flow at the PAV inlet, using a 2D model of tritium transport in the PAV channels developed with the OpenFOAM software, which implements a custom solver and tailored boundary conditions at the liquid–solid and solid–vacuum interfaces. Such CFD model is used to derive a correlation for the local Sherwood number in the entrance region as a function of the dimensionless Reynolds and Schmidt numbers and of the axial coordinate along the channel.

The new correlation for the entrance region, used together with the existing formulation for the fully developed flow region, is implemented in the system level GETTHEM code to allow the model validation against experimental data from the PAV mock up tests performed at TRIEX II and, in perspective, to support a more reliable sizing analysis of the PAV units for the DEMO WCLL PbLi circuits.

F4E_D_3EWG3M - F4E–EUROfusion joint programme for development Breeding Blanket Technologies

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

Following the 2017 technical and programmatic assessment of European breeding blanket activities, Fusion for Energy (F4E) and EUROfusion strengthened their collaboration to streamline the Test Blanket Module (TBM) and Breeding Blanket (BB) programmes, ensuring coherence with the European fusion roadmap objectives. The Water-Cooled Lithium Lead (WCLL) and the Helium-cooled Pebble Bed (HCPB) concepts remain relevant breeding blanket options for DEMO, whose reliable operation requires progressive experimental validation to de-risk technologies prior to DEMO deployment.

A central aim of the joint programme is to generate Return of Experience (RoX) from dedicated testing activities to support both TBM qualification and DEMO-relevant BB design. Collaborative R&D focuses on experimental validation of key technologies under representative operating conditions, including thermo-hydraulic performance, tritium management and system integration. Pilot-plant-oriented experiments are used to assess component and subsystem performance, providing essential data to reduce uncertainties and inform design choices.

The programme is structured around four main pillars, system modelling, technology R&D, safety, and materials qualification, with particular emphasis on the structural qualification of EUROFER97 reduced-activation ferritic-martensitic steel, manufacturing techniques, and operational limits. Through the pooling of expertise and experimental infrastructure, F4E and EUROfusion provide coherent guidance for DEMO BB design, focusing on the verification of high-level requirements such as tritium self-sufficiency, tritium confinement, structural integrity, and power extraction.

This paper presents an overview of collaborative R&D activities carried out during 2021–2025, highlighting how testing campaigns and RoX contribute to de-risking the TBM and DEMO blanket technologies and, in particular, supporting the consolidation of the breeding blankets design for DEMO.

#14048

CFD Modelling of Tritium Transport and Permeation Barriers in the EU-DEMO WCLL Breeding Blanket under MHD Conditions

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

In the EU-DEMO Water-Cooled Lithium Lead (WCLL) breeding blanket concept, controlling tritium permeation from the breeder material (PbLi) to the cooling water through the structural steel EUROFER97 represents a critical challenge for ensuring both reactor safety and tritium self-sufficiency. To mitigate tritium losses, permeation barrier (PB) materials are being actively investigated either as thin coatings or as bulk resistive interlayers integrated in structural components. In this context, the development of advanced numerical tools capable of accurately describing the coupled thermo-hydraulic, magnetohydrodynamic (MHD), and mass-transfer phenomena governing tritium transport is of paramount importance.

In this work, a Computational Fluid Dynamics (CFD) model is developed within the open-source OpenFOAM framework to investigate tritium transport in a simplified geometry representative of WCLL blanket structures. The configuration consists of a double-walled tube (DWT) with an intermediate permeation barrier layer, surrounded by electrically conducting liquid PbLi and partially enclosed by solid walls. The numerical model builds upon an existing solver for transient buoyant turbulent flows with conjugate heat transfer and incorporates new capabilities to account for tritium mass transport under MHD conditions. These developments include dedicated boundary conditions to capture concentration discontinuities at interfaces between materials with different solubilities, as well as surface dissociation and recombination processes at fluid–solid interfaces.

The model is first validated against benchmark problems available in the literature, including tritium permeation through composite membranes and MHD flows in conducting ducts, showing very good agreement with reference data. The validated framework is then applied to assess the effectiveness of several candidate PB materials. Three materials are considered: iron, currently adopted as a reference design option, and two ceramic alternatives, Al_2O_3 and B_4C. Simulations are performed under steady-state conditions with spatially dependent heat and tritium source terms representative of blanket operating conditions.

The results indicate that the use of advanced PB materials can reduce tritium permeation losses to the coolant by up to six orders of magnitude compared with the reference configuration. Furthermore, the analysis identifies the dominant transport mechanisms governing tritium migration within the system and evaluates the sensitivity of the results to key model parameters. Overall, the proposed modelling framework provides a robust predictive tool for supporting tritium management strategies and guiding the design and safety optimization of breeding blanket systems in future fusion reactors.

#14070

Cryogenical exposed electrofromed samples and viewing window for the ITER Fueling Pellet Injector

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

The ITER Pellet Injector System and internal fuel cycle components associated with it will be interacting with hydrogen isotopes and in some instances be providing a containment barrier. Traditional joining methods include brazing and sealing with polymer components will no longer be viable due to the transmutative properties of hydrogen isotopes. Furthermore, traditional welding methods require significant introduction of heat, which can lead to serious deformation and loss of strength, especially between dissimilar metals.

One of the main paths toward reconciliations of these issues is to use the electroforming process as a manufacturing technique replacing brazing, welding, and elastomer sealing. There are several methods of electroforming, but all utilize the electrodepostion process. This research examines several different electrodepostion processes. The four processes examined include electroforming SS316L to SS316L, SS316l to OFHC copper, OFHC copper to 1/4" VCR fittings, and OFHC Copper in a multistep, masking process. All of the components were subjected to 77K cryogenic conditions, pressuized with helium gas from one side, and evacuated to ultra high vacuum levels from the other side to determine leak tightness. Finally, a novel all metal sealed sapphire vacuum viewing window was also subjected to helium pressurization and ultra high evacuation at 77 K. The results of the testing of the various electroformed samples and the sapphire viewing windows are to be presented.

#14071

Progress on the Fusion Innovation Research Enginer Blanket Collaborative on Test Facilities (FIRE-BCTF)

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Oral

Abstract

As part of the US fusion technology ecosystem, the Department of Energy Fusion Energy Sciences program has formed a series of "Fusion Innovation Research Engine (FIRE)" Collaboratives, which are teams working across the country to bridge the technological gap between fundamental science and application to deploying fusion energy. The Oak Ridge National Laboratory (ORNL)-led FIRE Blanket Collaborative on Test Facilities (FIRE-BCTF) is a Collaborative focused on blanket technology and building test facilities to help deploy pre-competitive technology to be used across the fusion community. The project targets tritium breeding blanket challenges with heat removal, magnetohydrodynamic (MHD) effects in liquid breeder materials, material compatibility, and tritium extraction while working on code validation and integration throughout the duration of the project.

This Collaborative, while working with other FIRE Collaboratives to gather and share important findings, aims to advance blanket technology by building necessary component-scale facilities and collaborating with key partners to run experiments studying molten salt (FLiBe) blankets, lead-lithium (PbLi) blankets, helium cooling strategies, tritium extraction systems, and material compatibility in prototypic and multi-effect testing environments. A FLiBe molten salt forced convection loop is being designed and built at ORNL to study flow behavior, compatiblity between dissimilar fusion-relevant materials, heat removal, and tritium extraction systems using deuterium or hydrogen as a surrogate. To study PbLi blanket technology, this Collaborative is working with team members at Kyoto Fusioneering and the UKAEA to examine material compatibility for an alumized test article and test MHD diagnostics first on Kyoto Fusioneering's UNITY-1 facility and then deploy a prototype on the UKAEA CHIMERA facility to examine different MHD effects, heat transfer, and mass transfer in a transient magnetic field environment. To study helium cooling at prototypic flow rates and heat fluxes, this project will coordinate with the high-heat flux testing facility being constructed jointly by Type One Energy, the University of Tennessee Knoxville, and ORNL to run experiments on a test article.

To support these activities and the general needs of the fusion blanket technology community, material compatiblity experiments are performed across multiple institutions in the project to test corrosion of materials in FLiBe and PbLi and explore isotopically-tailored material solutions to achieving low-activation goals.

This work highlights the overall progrss of this project and how it integrates within the FIRE Collaborative efforts and the US fusion technology community as a whole.

Development of a Torus Vacuum Pumping System for a Volumetric Neutron Source

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Topics

I. Fuel Cycle and Breeding Blankets

Communication type

Poster

Abstract

To qualify in-vessel components for a fusion power plant at a technical readiness level (TRL) of 7, they must be operated in a relevant environment. A beam-driven tokamak with $Q < 1$ is proposed as a volumetric neutron source, providing the necessary environment and conditions representative of a fusion power plant. This concept is currently being developed as part of a conceptual design study within EUROfusion. The objective is to design a device capable of achieving fusion power levels of up to 40 MW in continuous operation with deuterium–tritium fuel. Owing to its compact design, the device can also provide a neutron wall load of 0.5 $MW.m^{-2}$ and an end-of-life neutron fluence of 50 dpa with tritium supplied from external sources.

The reference design features a single-null divertor configuration and incorporates four tangential positive-ion neutral beam injectors as well as an electron cyclotron heating system. Plasma fuelling is provided by pellet injection, while the injection of krypton as a seeding gas ensures operation in the detached regime. All injected gases must be continuously removed from the vacuum vessel (VV) by an appropriately designed pumping system.

The torus employs a multi-stage pumping concept consisting of torus cryosorption pumps (TCPs) combined with roots and scroll pumps. From a vacuum technology perspective, the cryopumps should be installed as close as possible to the VV. In total, eight TCPs are foreseen for the VV and are operated intermittently in a cyclic mode, alternating between pumping and regeneration phases, to guarantee continuous exhaust capability. The design is based on the ITER concept featuring an integrated high-vacuum valve. To increase pumping time without exceeding the hydrogen inventory limit, an additional volume is accommodated in the port cell and connected to each TCP. Further gas compression is achieved by an appropriate combination of roots and scroll pumps located in the tritium plant. This plant is housed in a building adjacent to the tokamak building, with the pumps connected via a dedicated inter-building piping system.

This paper presents the currently considered integrated concept of the torus pumping system, the rationale behind the design choices, and the design status of the vacuum system.