Loss of cryostat vacuum due to large ingress of helium gas

Matjaž LESKOVAR (Jožef Stefan Institute)
Janez KOKALJ (Jožef Stefan Institute)
Rok KRPAN (Jožef Stefan Institute)
Mitja URŠIČ (Jožef Stefan Institute)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

The presented study investigates a postulated accident scenario in the DEMO fusion reactor in which the cryostat is filled with helium gas due to a break in the cryogenic system used to cool the superconducting magnets. The MELCOR for Fusion code is applied to simulate the transient response and to evaluate pressure and temperature conditions in the enclosures between the vacuum vessel and the cryostat. Because the DEMO cryostat is a thin-walled structure, excessive pressurization could lead to buckling and potential collapse. Therefore, mitigation measures are assessed to ensure pressures remain below the cryostat design limit.

Eight accident scenarios were analysed to identify bounding conditions. Each scenario assumes a double guillotine break in only one cryogenic subsystem at a time. For magnet cooling systems, breaks were postulated below the central solenoid, poloidal field, or toroidal field coils, in piping between feeders and coils. For the thermal shield cooling system, simultaneous breaks of both pipes above the thermal shields were considered. Intact cryogenic subsystems were assumed either passive, representing a loss of off-site power, or active, to assess their influence. A sensitivity analysis was conducted to identify parameters most affecting cryostat overpressure.

Key results are presented for multiple locations in the DEMO tokamak building, including pressure, pressure differences, helium mass flow rate, cumulative released helium mass, void fraction, and temperature. The simulations indicate that pressure differences between the inside and outside of the cryostat are only slightly reduced in active scenarios compared with passive ones. The scenario with a break below the central solenoid coil is found to be bounding. The most influential parameters on cryostat overpressure are the cryostat atmosphere heat transfer coefficient, helium inventory and free volume, and the number and size of rupture discs. Conversely, the cryostat wall heat transfer coefficient, gallery leakage, magnets liquid helium heat transfer coefficient, break size, and rupture disc flow path length have negligible influence.

To maintain the pressure difference across the cryostat below the design limit of 9 kPa, it is recommended to install 14 rupture discs with a diameter of 0.1 m, opening at an overpressure of 5 kPa, distributed across the 16 cryostat sectors, with two backup locations available. In all scenarios, the bioshield safety assessment pressure limit of 200 kPa was not exceeded. The minimum temperature in the compartments outside the cryostat was approximately 265 K.

Application of ASME Section VIII to the Manufacture of ITER Nuclear Pressure Equipment under Module H1

Yugyeong KIM (ITER organization)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

ITER is a nuclear basic facility in France, where the manufacturing of nuclear pressure equipment must comply with French regulations. In this regulatory framework, codes and standards serve as tools to meet the essential safety requirements of ESPN order and PED. Therefore, manufacturer is solely responsible for demonstrating how the essential safety requirements are fulfilled by applying the chosen codes and standards for their conformity assessment, unless harmonized standards are used.

Under ITER's module H1, several nuclear vessels are being manufactured using the ASME section VIII division 2, a widely recognized code for pressure vessel. Despites its global acceptance in field, a detail gap analysis is mandatory to address the limitations of the ASME code in fulfilling the essential safety requirements specific to the French regulatory context.

This paper presents practical applications of ASME section VIII division 2 with compensatory measures for conformity assessment under ITER module H1. It outlines the key regulatory concepts, identifies essential safety requirements not fully covered by ASME from design, procurement, fabrication, and transversal perspectives, and explains how ITER has addressed these gaps across various types of nuclear pressure equipment.

The use of international standards to meet the requirements of French nuclear regulations in the manufacture of nuclear pressure vessels represents a critical challenge in the context of today's globalized nuclear industry supply chain. This paper aims to provide valuable insights into this important and timely topic.

Explosion safety for the Wendelstein 7-X central exhaust system

Sven DEGENKOLBE (Max Planck Institute for Plasma Physics)
Erik SCHARFF (Max Planck Institute for Plasma Physics)
Torsten BRÄUER (Max Planck Institute for Plasma Physics)
Lars KOCH (Max Planck Institute for Plasma Physics)
Rüdiger KRAMPITZ (Max Planck Institute for Plasma Physics)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

In the Wendelstein 7- X torus hall the exhaust from vacuum pumps, pressure relief valves and gas extraction systems must not be released into the torus hall. Instead, the systems must be connected to the central exhaust system. This exhaust system ensures a safe transport of the exhaust gas release into the atmosphere at a suitable location.
Depending on the state of operation of Wendelstein 7-X the exhaust gas may contain flammable or even explosive gases such as hydrogen and diborane as well as plain air.
The vacuum systems will release larger amounts of hydrogen into the exhaust system during plasma operation, glow discharge cleaning, boronization, cryopump regeneration and steady state pellet injector operation. If the concentration of the flammable gas exceeds the lower flammability limit, explosion protection measures must be installed. For the European Union the ATEX directives describe the minimum safety requirements for equipment (ATEX 114 (2014/34/EU)) and workplaces (ATEX 153 (1999/92/EG)) used in explosive atmospheres.
The poster will present the explosion-safe design of the central exhaust system. Scenarios involving the release of large quantities of flammable gases into the exhaust system will be analyzed in terms of explosion risks. Finally, the measures for explosion protection of non- ATEX certified components connected to the central

Preliminary accident analysis of the LOVA for the Korean innovative fusion reactor

MOON SUNGBO (Korea Institute of Fusion Energy (KFE))
Kim MYUNGHO (Korea Institute of Fusion Energy (KFE))
Kang GYOUNG-O (Korea Institute of Fusion Energy (KFE))
Kim IN-CHUNG (Korea Institute of Fusion Energy (KFE))
Her NAMIL (Korea Institute of Fusion Energy (KFE))

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

Preliminary accident analysis is conducted for representative Loss of Vacuum Accident (LOVA) conditions in the Korean innovative fusion reactor at the concept design stage. To address the limited design maturity, the study adopts a bounding, conservative methodology to establish an initial safety basis and to identify the primary contributors to vacuum vessel (VV) pressurization. Two initiating events are examined. First, air ingress into the vacuum vessel is postulated through a representative penetration breach, and a set of break areas is selected for parametric study. Second, Vacuum vessel pressurization due to in-vessel helium coolant ingress is evaluated following a postulated failure of helium-cooled in-vessel components, with key source-term characteristics treated parametrically.

A system-level, lumped-parameter model is developed to compute the vacuum vessel pressure transient, including the pressurization rate and peak pressure, for both air-ingress and helium-ingress cases. On this basis, the study further proposes a concept-level design and sizing methodology for a Vacuum Vessel Pressure Suppression System (VVPSS) tailored to the Korean innovative fusion reactor. The influence of major VVPSS design variables—such as suppression tank volume, relief line capacity, opening setpoints and logic, and effective suppression/condensation performance—is quantified and directly linked to the calculated VV pressure response.

The combined accident and mitigation assessment provides an integrated framework for early design iterations, enabling trade-off studies between confinement margins and mitigation capacity and informing preliminary performance targets for VV pressure control. The study also identifies priorities for follow-on work as the design evolves, including improved representation of penetration and port-cell coupling, more realistic helium-loop isolation and discharge modeling, and code-to-code benchmarking with higher-fidelity thermal-hydraulic and transport analyses.

#12832

Identifying Dominant Radiological Risk Drivers in Fusion Power Plant Source Terms

El Mehdi GHARDI (Nuclear Futures Institute, Bangor University)
Patrick WHITE (Clean Air Task Force)
Laurence WILLIAMS (Nuclear Futures Institute, Bangor University)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

Fusion power plants (FPPs) are expected to manage and process radioactive inventories arising from tritium and activated materials from in-vessel components. In postulated accident scenarios, the transport and dispersion of released radionuclides to the atmosphere, and the resulting radiological impact, are governed by a combination of design-specific and operational factors that defines the concentration and the composition of radionuclides in the source terms. This work investigates the influence of the key parameters on the effective dose using atmospheric dispersion modelling. Representative source terms are constructed based on radionuclides of concern identified from compiled radioactive inventory datasets for a range of FPP concepts. The analysis considers variations in release magnitude, radionuclide composition, and meteorological conditions to evaluate their relative contribution to off-site dose. By systematically exploring the sensitivity of atmospheric dispersion parameters to fusion-specific design and operational factors, this study aims to identify the dominant drivers of radiological consequences in accident scenarios. The results provide insight into how inventory composition and plant operation influence doses, and support the development of graded, technology-informed approaches to fusion safety.

#12836

Effect of Water Moderation on RAFM Steel Activation: a radially resolved analysis of the DEMO WCLL blanket

Virginie LOMBARDI (Sapienza University of Rome, Department of Electrical and Energy Engineering (DIEE), Roma)
Priti KANTH (United Kingdom Atomic Energy Authority, Culham Campus,Abingdon, Oxfordshire)
Giada GANDOLFO (ENEA C.R.Casaccia,ENEA,Roma)
Fabio MORO (ENEA C.R. Frascati, ENEA, Frascati, Roma)
Matteo D'ONORIO (Sapienza University of Rome, Department of Electrical and Energy Engineering (DIEE), Roma)
Gianfranco CARUSO (Sapienza University of Rome, Department of Electrical and Energy Engineering (DIEE), Roma)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

Neutron-induced activation of structural materials is the dominant source of radioactive waste in a tokamak, and it could represent a key challenge for fusion reactor decommissioning. Reduced Activation Ferritic Martensitic (RAFM) steels have been developed to mitigate this issue by minimising the production of intermediate-level waste (ILW). Although several studies have assessed the overall waste inventory expected from DEMO for different blanket concepts, a detailed evaluation of activation as a function of the radial distance from the plasma remains limited. Such information is essential for dose rate assessment and for optimising dismantling procedures and waste disposal strategies.

This work investigates the radial distribution of steel activation in the Water-Cooled Lithium-Lead (WCLL) blanket, with particular emphasis on the effect of the thermal neutron peak generated by water moderation. Neutron spectra were calculated with MCNP on a 25-point radial mesh within an equatorial outboard breeding unit, from the first wall (FW) to the back supporting structure (BSS), using the heterogeneous 2022 DEMO model with helical cooling tubes. The resulting spectra have been used as input for the FISPACT-II inventory code to perform activation calculations.

The reference DEMO irradiation scenario consists of two phases: an initial testing phase of approximately five years, after which a blanket replacement is foreseen, followed by a longer operational phase of nearly fifteen years. For this study, blanket materials were irradiated in FISPACT-II according to both phases, in order to quantify differences in activation build-up and their implications for waste classification.

A comparative analysis has been carried out for different RAFM steels, such as EUROFER97 and F82H, and for different nuclear data libraries (TENDL and JEFF), enabling an evaluation of the sensitivity of activation predictions to the selected nuclear data.

Results confirm the shielding effect of water for components located further from the plasma. However, the associated thermal neutron peak locally enhances the production of radionuclides with high thermal activation cross-sections, such as Fe-55, Ni-63 and C-14. In regions close to the water manifolds, this effect leads to activity levels exceeding the UK low-level waste (LLW) limits even 50 and 100 years after shutdown.

These findings highlight a non-negligible activation hotspot intrinsic to the WCLL design and underline the importance of radially resolved activation analyses for accurate waste classification and decommissioning planning in future fusion reactors.

Development of a One-Dimensional Safety Analysis Code for the Divertor During a Loss of Flow Accident

Ceyda ELDEM (Pohang University of Science and Technology (POSTECH))
Zumaro ZUMARO (Pohang University of Science and Technology (POSTECH))
Yohan LEE (Pohang University of Science and Technology (POSTECH))
UngJin NA (Pohang University of Science and Technology (POSTECH))
Gyudong LEE (Pohang University of Science and Technology (POSTECH))
Muhammad Aghist FITRONY (Pohang University of Science and Technology (POSTECH))
Yongnam LEE (Pohang University of Science and Technology (POSTECH))
HangJin JO (Pohang University of Science and Technology (POSTECH))

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

Loss-of-flow accident (LOFA) is recognized as a relevant design-basis accident of a fusion reactor, particularly due to its potential impact on in-vessel components exposed to high heat fluxes. In the divertor cooling system, the consequences of LOFA strongly depend on the time available for plasma shutdown. This time margin is governed by the transient coolant behavior following pump coastdown. While widely used system codes have been applied to such scenarios, their generic heat structures and correlations may not adequately resolve the short time scales and high heat-flux conditions characteristic of divertor cooling channels. In this study, a dedicated one-dimensional safety analysis code is developed to analyze divertor cooling channels under LOFA conditions. Unlike conventional system codes, the model employs heat structures and correlations tailored to divertor applications and is designed to provide flexibility with respect to cooling channel geometry, mass flow rate, and hydraulic boundary conditions. The code focuses on the transient period following pump coastdown up to a predefined protection threshold, with particular attention to the time to burnout (TBO), which represents a key safety-relevant metric for plasma shutdown and component integrity. The modeling framework enables detailed parametric and sensitivity analyses that are difficult to achieve experimentally due to the cost and complexity of high-heat-flux testing. Preliminary results indicate that TBO is sensitive to multiple parameters, including pump coastdown behavior, operating conditions, and channel characteristics, highlighting the importance of accurately capturing transient thermal-hydraulic responses. The modular framework allows further refinement as additional data or improved correlations become available. The developed code provides a flexible analysis tool for divertor LOFA studies, supporting systematic investigation of parameters influencing TBO and CHF margin. Such analyses can assist in the interpretation of accident scenarios and contribute to the development of design and accident criteria for high-heat-flux components in fusion reactors.

System-Level Multi-Fluid Modelling of the ITER Water-Cooled Lead-Lithium TBM with MELCOR-TMAP

Matteo D'ONORIO (Sapienza University of Rome)
Tommaso GLINGLER (Sapienza University of Rome)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

System-level codes are extensively used to support safety and transient analyses in fusion systems. However, legacy MELCOR models rely on a single-fluid framework, which prevents the simultaneous representation of coolant and liquid-metal breeder flows and limits the applicability to multi-fluid breeding blanket systems.
In previous breeding blanket MELCOR models, the liquid metal breeder was represented as a static solid. This assumption was acceptable when liquid-metal advection could be neglected, but it prevented investigation of transients in which PbLi flow could play a significant role and introduced modelling constraints related to the representation of breeder inventory and geometry.
Recent developments in MELCOR-TMAP enable multi-fluid simulations within a single input deck. Building on a previously developed MELCOR model, a new system-level model of the ITER Water-Cooled Lead-Lithium Test Blanket Module has been developed, treating both water and PbLi as working fluids. While the Water-Cooling System representation has been retained from the previous model, the TBM model has been substantially redesigned to explicitly include a PbLi fluid domain. In addition, a dedicated LiPb loop model feeding the breeder units has been implemented. Explicitly treating PbLi as a working fluid in the developed TBM model ensures a physically consistent representation of the system design and significantly expands the range of analyzable operating and accidental scenarios.
The model is first evaluated under reference operating conditions by comparing key parameters, such as coolant pressure and component temperatures, with design values. A Loss of Flow Accident is then investigated as a demonstration case. The event is caused by the spurious closure of the SIV-1 located at the TBM outlet during an ITER plasma burn phase. The study shows that the multi-fluid approach enables evaluation of coupled thermo-hydraulic behaviour and supports assessment of pressure and EUROFER temperature design limits, representing a first step toward analysing PbLi-related accident scenarios previously inaccessible with legacy models.

From Surface to Bulk: The MELTE Project for Deep Detritiation of Fusion Components

Florian PRIESTER (Karlsruhe Institute of Technology (KIT))
Dominic BATZLER (Karlsruhe Institute of Technology (KIT))
David HILLESHEIMER (Karlsruhe Institute of Technology (KIT))
Robin GRÖSSLE (Karlsruhe Institute of Technology (KIT))
Alexander MARSTELLER (Karlsruhe Institute of Technology (KIT))
Simon NIEMES (Karlsruhe Institute of Technology (KIT))
Marco RÖLLIG (Karlsruhe Institute of Technology (KIT))
Marius SCHAUFELBERGER (Karlsruhe Institute of Technology (KIT))
Michael STURM (Karlsruhe Institute of Technology (KIT))
Stefan WELTE (Karlsruhe Institute of Technology (KIT))

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

A so far not addressed issue arising from the operation of tritium laboratories and tritium processing plants concerns the disposal of parts and potential recycling of materials after end-of-service. Tritium on surfaces and in the metal lattice of tritium facing components cannot be removed efficiently enough with current technologies. Free release limits and the acceptance limits by nuclear waste disposal facilities often restricts disposal of components contaminated with tritium and increase related costs, up to a point where disposal is no longer a viable option.

With an emerging fusion-energy economy, and hence a multitude of different sites handling tritium in significant amounts, the challenge of disposal now becomes a pressing matter imposing significant logistical and economic burdens. Consequently, TLK is expanding its current research scope from surface-focused decontamination techniques to include bulk decontamination methods. This includes research efforts for short-term activities like maintenance and upgrades as well as deep detritiation.

By designing a facility capable of melting stainless steel components, the TLK aims to achieve deep detritiation of bulk materials. This thermal treatment is expected to efficiently release tritium trapped deep in the metallic bulk, effectively opening the waste-path to mid- or low-level regulatory standards with a best-case scenario of allowing the release of waste as non-radiative.

TLK is uniquely positioned to lead this effort. Having successfully verified the process on a small scale without tritium, we are now transitioning to an active small-scale demonstration unit. The ultimate goal of the setup is a semi-industrial, 100 kg-scale batch process for decontamination of large components.

Equipped with a comprehensive suite of diagnostics, the proposed MELTE (Melting for Tritium Extraction) facility will provide the essential data and methodologies required to manage the future legacy of fusion research and fusion power generation, ultimately answering the end-of-service question and fostering broader public acceptance.

Experimental validation activities in support of the lithium fire protection strategy of the IFMIF-DONES facility

Gianluca D'OVIDIO (CIEMAT - National Fusion Laboratory)
Víctor GUTIÉRREZ (CIEMAT - National Fusion Laboratory)
Johan ANDERSON (RISE)
Paul OTXOTERENA (RISE)
Julián PATIÑO (CIEMAT - National Fusion Laboratory)
Marcelo ROLDÁN (CIEMAT - National Fusion Laboratory)
Teresa HERNÁNDEZ (CIEMAT - National Fusion Laboratory)
Alberto ARTACHO (IFMIF-DONES Spain Consortium)
Francisco MARTÍN-FUERTES (CIEMAT - National Fusion Laboratory)
Joaquín MOLLÁ (CIEMAT - National Fusion Laboratory)
A. DE CASTRO (CIEMAT, National Fusion Laboratory)
L. SERRADOR (CIEMAT - National Fusion Laboratory)
F. ARBEITER (Institute for Neutron Physics and Reactor Technology (INR), Karlsruhe Institute of Technology (KIT))
D. KLIMENKO (Institute for Neutron Physics and Reactor Technology (INR), Karlsruhe Institute of Technology (KIT))

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

In order to progress towards the final design of the IFMIF-DONES facility, it is imperative to demonstrate an acceptable risk of potential lithium fire scenarios and to provide the facility with the adequate safety measures for prevention and mitigation of these scenarios. In this context, experimental R&D activities are being developed to primarily demonstrate the safe use of dry air for room inertization, to characterize the lithium reaction products generated in a fire, to test the chemical compatibility of functional materials with lithium, and to identify the most suitable extinguisher for lithium fires.
Using dry air instead of inert gas in those rooms, housing lithium systems, would significantly simplify the ventilation system design, reduce the complexity of operational and maintenance activities and minimize the risk of anoxia for the workers within the facility. As demonstrated by the experiments conducted at CIEMAT, the probability of ignition occurring in static lithium pools, which could be representative of confined lithium leaks, is considered minimal in air at temperatures below 400°C and under low humidity conditions (RH

#13097

Preliminary accident analysis of loss of flow for the European DEMO using the Helium Cooled Pebble Bed blanket concept

Xue Zhou JIN (Karlsruhe Institute of Technology (KIT))

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

Deterministic accident analyses for the Design Basis Accident (DBA) and Design Extension Condition (DEC) analyses play a central role in the safety and licensing strategy of the European DEMO reactor. In this study, the selected postulated initial event (PIE) is the loss of flow in the primary cooling loop of the first wall (FW) and blanket structures because compressor or pump seizure (LOFA). The PIE is investigated for the Helium cooled Pebble Bed (HCPB) blanket concept according to the DEMO baseline 2017. MELCOR 1.8.6 for fusion is selected by the Safety Office in EUROfusion for performing accident analysis. A comprehensive DEMO model has been developed using MELCOR, comprising the HCPB breeding blanket (BB) system, the associated primary heat transfer system (PHTS), the purge gas loop, the Plasma Facing Unit (PFU) loops for the divertor, and other related systems in the tokamak building such as the vacuum vessel (VV), the VV pressure suppression system (VVPSS), the cryostat and cryostat space, and the gallery. The boundary conditions for the plasma behavior, inter-compartment leakage, venting, and other relevant processes are incorporated into the model.

For the DBA, the LOFA is postulated to occur in a BB loop of the PHTS by compressor seizure. The fast plasma shutdown (FPS) is triggered 3 s after the detection of LOFA based on the low mass flow rate signal, and it is followed by a mitigated plasma disruption. When the temperature of the affected FW reaches the defined threshold of 800 °C (TEF), the FW fails, which constitutes an aggravating in-vessel LOCA (Loss of Coolant Accident). In contrast, in the DEC scenario the FPS is assumed to fail, causing the plasma to continue burning until the FW reaches the TEF and subsequently fails. Due to He ingress into the VV, the plasma is terminated by an unmitigated plasma disruption. It is further assumed that once the temperature of the PFU monoblocks reaches the TEF, they will also fail. In such a case, water from the PFU loop will ingress into the VV. The pressurization of the VV will drive the mobilized radioactive inventories – tritium and dust in the VV, as well as ACP in the PFU loop – towards the connected systems within the tokamak building, from where they may eventually be released to the environment. Following the accident sequences, the thermohydraulic transients and releases will be assessed in an overall evaluation.

Operational Availability as a Bottleneck in Fusion Replication

Sophie BROEKERS (Eindhoven University of Technology)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

Fusion energy is often evaluated in terms of technical feasibility or projected cost competitiveness. However, the critical socio-economic challenge lies in the transition from commercial demonstration plants to early replication. This phase determines whether fusion stabilises as industrial infrastructure or remains a sequence of bespoke projects. During this interval, the stability of the safety case, licensing credibility, insurability, grid compatibility, and financeability become simultaneously binding constraints. Progress therefore depends not on average technical performance, but on whether operational risk can be credibly bounded across institutional domains.

This research identifies operational availability as a cross-domain coupling variable linking engineering design, regulatory acceptance, insurance assessment, and financial viability. In most techno-economic studies, availability is represented as a scalar capacity factor assumption. In practice, availability is an emergent property of the reliability–availability–maintainability–inspectability (RAMI) architecture, shaped by maintainability design, component lifetimes, inspection regimes, remote handling logistics, outage clustering, and recirculating power constraints. Many of these drivers are largely fixed during demonstration design, yet they are evaluated under more conservative criteria during first-of-a-kind licensing and project finance.

The analysis reframes availability as a stochastic distribution rather than a mean value. Regulators evaluate safety case stability under outage conditions. Insurers assess exposure to loss under repair uncertainty. Lenders examine downside cash-flow resilience under P90 availability scenarios. The relevant object therefore becomes the tail of the availability distribution rather than its average. High-performance operating regimes may increase stress dispersion and repair variance, expanding financial downside risk even if nominal output improves.

The causal chain linking early design choices, availability tail behaviour, and subsequent financeability and licensing thresholds is identified. By treating availability as a socio-technical interface parameter, the study clarifies why demonstration success does not automatically translate into scalable deployment. The resulting decision-relevant framework links safety requirements, operational evidence, and financial acceptance, providing structured guidance for derisking strategies in early commercial fusion.

Effect of Packing Architecture on H?–O? Recombination Efficiency in a Trickle-Bed Reactor under Distributed Hydrogen Injection

Gheorghe BULUBASA (National Institute of Research & Development for Cryogenic & Isotopic Technologies Ramnicu Valcea)
Maria CRACIUN (National Institute of Research & Development for Cryogenic & Isotopic Technologies Ramnicu Valcea)
Alina NICULESCU (National Institute of Research & Development for Cryogenic & Isotopic Technologies Ramnicu Valcea)
George ANA (National Institute of Research & Development for Cryogenic & Isotopic Technologies Ramnicu Valcea)
Iuliana STEFAN (National Institute of Research & Development for Cryogenic & Isotopic Technologies Ramnicu Valcea)
Ciprian BUCUR (National Institute of Research & Development for Cryogenic & Isotopic Technologies Ramnicu Valcea)
Anisia BORNEA (National Institute of Research & Development for Cryogenic & Isotopic Technologies Ramnicu Valcea)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

Hydrogen–oxygen recombination constitutes an essential safety function in fusion-related facilities, where hydrogen isotopes, including protium, deuterium, and tritium, may be involved during operation, maintenance, or waste treatment and conditioning activities. In particular, the management of hydrogen isotopes containing off-gas streams is critical to limit explosion hazards and to prevent the uncontrolled release of tritium into the environment. Catalytic recombiners are therefore regarded as a key mitigation technology, enabling the conversion to water of hydrogen isotopes in oxygen-containing atmospheres under a wide range of operating conditions. A detailed understanding of how reactor design and internal packing architecture influence recombination behavior is consequently of direct relevance for improving the safety, environmental compatibility, and robustness of fusion technology systems.

In this work, the recombination performance of a trickle-bed reactor (TBR) operated under co-current downflow of air with distributed hydrogen injection is investigated, focusing on the role of internal packing architecture. Two different packing configurations are considered: an ordered catalytic packing, and a randomly packed but axially stratified configuration combining hydrophobic catalyst particles with hydrophilic stainless-steel elements.

The experimental campaign is conducted at hydrogen concentrations in air ranging from 1 to 4 vol.%, superficial gas velocities between 1.5 and 10 cm s^{-1}, and over a controlled range of bed temperatures selected to limit changes in liquid holdup and phase behavior. Hydrogen is introduced through multiple axial injection points to reproduce non-uniform feed conditions that may arise in practical safety systems. Reactor performance is assessed in terms of hydrogen recombination efficiency, and indicators of operational stability.

In both packing configurations, the same amount of platinum-based catalytic material is employed, ensuring that any observed differences in reactor behavior can be attributed to packing architecture rather than to intrinsic catalytic properties. This controlled comparison aims to clarify the role of functional packing design in enhancing the safety and reliability of hydrogen recombination systems relevant to fusion technology and associated waste management applications.

NuclearVerse: A comprehensive VR toolchain for Fusion Maintenance

Martín GORRIARAN (IDOM | Consulting, Engineering and Architecture)
Arnau GARCÍA (IDOM | Consulting, Engineering and Architecture)
Daniel MOYA (IDOM | Consulting, Engineering and Architecture)
Javier LEBOREIRO (IDOM | Consulting, Engineering and Architecture)
Enric BARGALLÒ (IDOM | Consulting, Engineering and Architecture)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

Maintenance in fusion facilities is constrained by extreme radiological fields, activation of materials, tight access envelopes, and complex, interdependent assemblies—conditions that elevate execution risk and complicate commissioning. A comprehensive Virtual Reality (VR) toolchain is presented, enabling engineering teams to plan, rehearse, and validate maintenance procedures before physical realization, with the dual objective of minimizing worker dose and reducing execution uncertainty.

The platform supports (i) scripted, repeatable rehearsal of disassembly and handling sequences with realistic physics; (ii) high precision in scene metrology for clearance and reach analyses; and (iii) automated route validation with clash/conflict detection to verify access, lifting paths, and tool envelopes. Embedded ergonomics assessment provides reachability and posture evaluation, envelope and clearance checks, and support for layout and sequencing decisions that mitigate operator workload and musculoskeletal risk. These capabilities surface geometric interferences, procedural bottlenecks, and ergonomics issues early, enabling design for maintainability trade offs and de risked commissioning. For industrial use, the system provides annotated procedures, versioned task scripts, and traceable rehearsal outcomes (videos, metrics, and reports) to demonstrate readiness.

A key contribution is a local, air gapped AI assistant that couples large language models with retrieval augmented generation over site specific document sets—component inventories, maintenance procedures, ergonomics limits, and regulatory constraints. This delivers context aware guidance inside VR while preserving data privacy and eliminating reliance on external connectivity.

Applied to fusion maintenance scenarios, the toolchain has demonstrated reductions in task variability and on site duration, with early detection of access conflicts that limit rework and support adherence to ALARA principles, and a potential of including Remote Handling tasks in the tool. Beyond fusion, the approach is transferable to other high hazard or access constrained environments (particle accelerators, chemical plants, space ground segments), offering a scalable pathway to safer, more efficient maintenance.

#13280

Sensitivity of shutdown dose rate and waste classification predictions to evaluated nuclear data in the IFMIF-DONES test cell

Dilmurod TUYMURODOV (Laboratory of Nuclear Installation Safety, Lithuanian Energy Institute, Breslaujos Str. 3, LT-44403 Kaunas, Lithuania)
Gediminas STANKUNAS (Laboratory of Nuclear Installation Safety, Lithuanian Energy Institute, Breslaujos Str. 3, LT-44403 Kaunas, Lithuania)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

Planning maintenance interventions and managing radioactive waste at IFMIF-DONES requires reliable predictions of shutdown dose rates and radionuclide inventories in activated structural components. The deuterium lithium neutron source produces a spectrum extending to approximately 55 MeV, well above the 14 MeV peak of DT fusion devices, entering an energy range where evaluated cross section libraries show larger mutual discrepancies and where covariance information is often incomplete. The extent to which these nuclear data differences translate into variations in predicted dose rates at maintenance access points or into shifts in waste classification timelines for irradiated components has not been quantified for DONES in an integrated manner.

This study addresses that question using MCNP6 coupled to FISPACT-II and the rigorous two-step (R2S) method, applied to a model of the DONES test cell and High Flux Test Module region under nominal beam conditions. Neutron flux spectra are tallied on a spatial mesh and collapsed for inventory calculations. Activation inventories, decay photon source terms, and shutdown dose rate distributions are evaluated at cooling times relevant to hands on maintenance (hours to weeks) and waste disposal (years to decades). The calculation sequence is repeated with recently released transport libraries (FENDL-3.2c, JEFF-4, ENDF/B-VIII.1, JENDL-5) and activation datasets (TENDL-2019, EAF-2010) to isolate the contribution of each data selection to variations in safety relevant quantities.

For each material and irradiation position the dominant radionuclide production pathways responsible for the largest inter library differences are identified through pathway analysis. Results are reported as percentage variation in shutdown dose rate at candidate maintenance positions, shift in time required to reach clearance or low-level waste thresholds, and a ranked list of the nuclear reactions whose evaluation differences drive these variations. The findings are intended to support nuclear data evaluation priorities within the EUROfusion Early Neutron Source programme.

Safety Approach and Preliminary Analyses in Support of the Design of the VNS Facility

Matteo D'ONORIO (Sapienza Unviersity of Rome, DIEE, Corso Vittorio Emanuele II 244, I-00186 Roma, Italy)

Martina MOLINARI (EUROfusion Consortium, Garching, Boltzmannstr. 2, D-85748, Germany; Sapienza Unviersity of Rome, DIEE, Corso Vittorio Emanuele II 244, I-00186 Roma, Italy)

Sandrine ROSANVALLON (CEA, IRFM Cadarache, France; DEMO Central Team, EUROfusion, Garching, Boltzmannstr. 2 85748 Germany)

Tommaso GLINGLER (Sapienza Unviersity of Rome, DIEE, Corso Vittorio Emanuele II 244, I-00186 Roma, Italy)

Joelle ELBEZ-UZAN (CEA, IRFM Cadarache, France; DEMO Central Team, EUROfusion, Garching, Boltzmannstr. 2 85748 Germany)

Sergio CIATTAGLIA (EUROfusion Consortium, Garching, Boltzmannstr. 2, D-85748, Germany)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

For the qualification of in-vessel components, the Volumetric Neutron Source (VNS) is currently under investigation as a quasi-continuous D–T plasma device that provides a 14 MeV neutron source. The VNS is expected to be relatively compact, with low fusion power and tritium consumption of less than 1 kg per full-power year. However, it will require a robust safety framework to ensure plant controllability and mitigate both radiological and industrial risks under normal and accidental conditions.

In line with fusion safety approaches, the design adopts principles such as Defence in Depth, ALARA, and passive safety, with the overarching objective of ensuring the confinement of radioactive inventory, minimizing workers' exposure, and limiting environmental releases. To verify the design's capability to fulfil these safety functions, preliminary accident analyses have been performed using the MELCOR system code.

The developed MELCOR model includes the Shielding Blanket, the water-cooled Breeding Blanket system, the Vacuum Vessel and its extensions, and the tokamak building.

Two scoping accident scenarios were analysed: an in-vessel Loss of Coolant Accident (LOCA) and an ex-vessel LOCA. Results indicate that, for the in-vessel LOCA, the Vacuum Vessel Pressure Suppression System (VVPSS) effectively limits pressure increase below design thresholds while confining contaminants and reducing releases. In the ex-vessel scenario, pressure relief flaps mitigate pressure buildup; however, steam migration towards galleries around the Upper Pipe Chase (UPC) leads to increased temperatures, detritiation system shutdown, and subsequent HTO release inside the tokamak building, followed by limited environmental leakage.

These analyses provide preliminary confirmation that the current VNS design can achieve the required safety functions while offering feedback for further design refinement.

Social Sustainability of Fusion Energy: Labor Risks of Global Fusion Supply Chain

Misa IWAMOTO (Kyushu University)
Naomi KERR (Keio University)
Alexander Ryota KEELEY (Kyushu University)
Daikichi SEKI (Kyushu University)
Shunsuke MANAGI (Kyushu University)
Shutaro TAKEDA (Keio University)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

Fusion energy is widely regarded as a future low-carbon electricity source. While research has focused on plasma performance, engineering feasibility, and economic viability, the social risks embedded in fusion's global supply chains remain largely unexamined, limiting understanding of its sustainability.

Given the projected pilot plant development in the late 2030s, assessing such risks prior to commercialization is increasingly important.

This study aims to comprehensively quantify labor-related social risks associated with the construction and operation of fusion power plants and to position fusion relative to other electricity generation technologies within a unified analytical framework.

The analysis considers a 1 GWe tokamak-type fusion power plant assumed to commence operation in the United Kingdom in 2035. The cost structure is based on the ARIES-ST design study by Najmabadi et al. (1992), adjusted to 2024 prices using a cumulative inflation factor of 2.25. A plant lifetime of 60 years and a discount rate of 5% are assumed, resulting in an estimated levelized cost of electricity of 112 USD/MWh. Primary equipment procurement is assumed to occur within the United Kingdom.

Methodologically, this study integrates a Multi-Regional Input–Output (MRIO) model based on the Eora Global Supply Chain Database (reference year: 2022; 189 countries; 14,839 sectors) with Social Life Cycle Assessment using the Social Hotspots Database, aligned with ISO 14040. The functional unit is 1 MWh of electricity. Social risks are measured in adjusted risk hours (arheq), covering 30 subcategories across five impact categories.

The results indicate that fusion energy generates 2.1 arheq/MWh, ranking fifth among twelve electricity technologies. Its social risk is slightly higher than that of large-scale solar PV (0.9 arheq/MWh), wind power (1.1 arheq/MWh), and fission (1.5 arheq/MWh). The category "Labor Rights and Decent Work" accounts for the largest share (42.5%) of total risk, with the "Wage" subcategory representing the dominant contributor (20% within the category).

Approximately 98.5% of total social risks occur outside the United Kingdom. Nigeria and the Democratic Republic of Congo are the top two contributors, each accounting for 15%. To examine the influence of procurement structure, a sensitivity analysis assuming an ITER-like international sourcing configuration results in a 24% increase in total social risk.

These findings demonstrate that the social sustainability of fusion energy is shaped by the structural dependence of material and component supply on countries characterized by low wages and institutional challenges. Accordingly, future fusion strategies must integrate responsible sourcing and supply chain governance alongside technological advancement.

Development of Innovative Modular Bioshield Using Open Cell Metal Foam Composites for Fusion Applications

Luigi CANDIDO (Kyoto Fusioneering UK Ltd)
Javier ALGUACIL (Universidad Nacional de Educación a Distancia (UNED), Madrid, Spain)
Paul BARRON (Kyoto Fusioneering Ltd.)
Colin BAUS (Kyoto Fusioneering UK Ltd)
Juan Pablo CATALAN (Universidad Nacional de Educación a Distancia (UNED), Madrid, Spain)
Chris ELSE (Kyoto Fusioneering UK Ltd)
Suneui LEE (Kyoto Fusioneering Ltd.)
Italo GODOY-MORISON (Kyoto Fusioneering Ltd.)
Mario OLIVER (Kyoto Fusioneering UK Ltd)
Ben RAEVES (Kyoto Fusioneering UK Ltd)
Alejandro REMIRO (Kyoto Fusioneering UK Ltd)
Patrick SAUVAN (Universidad Nacional de Educación a Distancia (UNED), Madrid, Spain)
Carli SMITH (Kyoto Fusioneering America Ltd.)
Jun TAKAMINE (Kyoto Fusioneering Ltd.)
Jack TAYLOR (Kyoto Fusioneering UK Ltd)
Andrew WILSON (Kyoto Fusioneering UK Ltd)
Satoshi KONISHI (Kyoto Fusioneering Ltd.)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

Future fusion power plants require bioshielding solutions that ensure effective neutron and gamma attenuation while addressing structural integrity, activation, and lifecycle performance. Kyoto Fusioneering has developed and assessed a modular bioshield concept based on an open-cell aluminium-based foam composite, investigated as an alternative to conventional concrete bioshields for fusion applications.

An initial literature review was used to survey and down-select candidate shielding approaches, leading to the selection of an open-cell aluminium foam composite as the reference concept. The selected architecture consists of an open-cell aluminium foam acting as a load-bearing porous skeleton, with the pore volume filled by a polymer matrix incorporating functional fillers to enhance neutron attenuation. This configuration enables a modular block-based bioshield design compatible with fusion plant integration requirements.

Then, comprehensive multiphysics analyses of the aluminium foam composite bioshield have been assessed. Neutronics analyses were performed to quantify neutron and gamma attenuation performance relative to conventional bioshield materials. Activation studies evaluated induced radioactivity, decay heat, and waste classification to assess long-term operational and decommissioning implications. Structural and thermo-mechanical analyses, including seismic loading, were conducted to examine the feasibility of modular block configurations under representative fusion plant conditions. In parallel, a lifecycle assessment (LCA) was carried out to evaluate environmental impacts and embedded carbon compared to traditional concrete-based solutions.

The results indicate that open-cell aluminium foam composite bioshields offer a promising modular solution for fusion facilities, with potential benefits in weight reduction, activation behaviour, structural adaptability, and lifecycle performance. This work demonstrates the feasibility of aluminium-based foam composites as an enabling technology for next-generation fusion bioshielding systems.

The expansion of fusion safety research activities at the Idaho National Laboratory to support the accelerated deployment of fusion energy

Pattrick CALDERONI (Idaho National Laboratory)
Adriaan RIET (Idaho National Laboratory)
Sai ZHANG (Idaho National Laboratory)
Casey ICENHOUR (Idaho National Laboratory)
Pierre-Clement SIMON (Idaho National Laboratory)
Guillaume GIUDICELLI (Idaho National Laboratory)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

In the last 40 years the Idaho National Laboratory (INL) has provided an essential contribution to DOE's office of Fusion Energy Science (FES) by providing safety analysis in support of the design of fusion systems. The Fusion Science and Technology roadmap in 2025 expanded the scientific mission of FES to support the design and successful operation of a US-based fusion pilot plant, with electricity generation planned for the end of the 2030s through the coordinated efforts of DOE labs, universities, and private industry. Building on the core capability established by the Fusion Safety Program, INL fusion research expanded its role in providing innovative research and development with focus on fusion nuclear technology. However, an important mission of the INL fusion program remains linked with plants safety assessment through coupled analytical and experimental activities. Modeling and simulation activities predict potential risks, hazards, and failure modes through system-level safety codes, complemented with multi-physics, high fidelity tools. Experiments are performed to validate models and demonstrate the performance of materials and components in prototypical environments.

The expanded fusion safety research activities integrate existing tools for fusion safety analysis with risk-informed design processes developed for fission plants and high-fidelity models to support fusion power-plant design. The following are examples of research outcomes that are discussed in this paper:
• expansion of MELCOR-TMAP to improve liquid metal blankets modeling and coupling to RAVEN, an open source software platform that facilitates and enhances a variety of model exploration, risk analyses and design optimizations for nuclear reactors, energy grids and other complex systems;
• open access, digital database of legacy documents complied by the INL Fusion Safety Program, including ARIES design studies;
• hazard analyses performed for Tokamak Energy's Spherical Tokamak (ST) concept and the Fusion Nuclear Science Facility concept;
• high-fidelity multiphysics tools based on the MOOSE computational framework including tritium transport and accountancy (TMAP8), plasma, scrape-off layer, and blanket physics (SALAMANDER), electromagnetics; and functional and structural fusion-relevant materials analysis;
• statistical and physics-based approaches and tools for reliability, availability, maintainability, and inspectability (RAMI) of fusion components.
• application of classical and dynamic risk assessment approaches and tools for fusion pilot plant designs and tritium handling system designs with the coupled use of TMAP8 and EMRALD, an INL-developed dynamic risk assessment tool;
• high-fidelity neutronic calculations within the MOOSE computational framework to assess tritium generation, materials activation, and fusion by-products.

Main outcomes from the German pilot project ReFus on the regulation of fusion facilities

Joachim HERB (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH)
Clemens HEITSCH (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH)
Manuela JOPEN (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH)
Michael KOWALIK (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH)
Michael KUND (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH)
Ilka PETERMANN (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH)
Kevin COHRS (TÜV Rheinland)
Jens-Uwe SCHMOLLACK (TÜV Rheinland)
Matthias ENGLERT (Öko-Institut, Institute for Applied Ecology)
Anna KOPP (Öko-Institut, Institute for Applied Ecology)
Christoph PISTNER (Öko-Institut, Institute for Applied Ecology)
Dirk RADLOFF (Karlsruhe Institute of Technology (KIT))
Xue Zhou JIN (Karlsruhe Institute of Technology (KIT))
Philipp SAUTER (Max-Planck-Institute for Comparative Public Law and International Law)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

The pilot project ReFus aimed to analyze the status-quo of the legal and technical regulations for fusion facilities in Germany, to develop a concept for integrating future fusion facilities or power plants into the existing legal framework, and to identify needs for further development. Additionally, inputs from start-up and industry companies supported the development of the regulatory framework, with their expectations and needs gathered through targeted interviews.

After examining the current legal framework, it was concluded that fusion facilities are covered by the German Radiation Protection Act: as facilities to produce ionizing radiation requiring an operation license and as plasma facilities requiring a construction license if capable of producing more than 1012 neutrons in one second. Also, a license for the use and handling of radioactive substances is needed. The creation of a special Fusion Energy Act (FEA) was considered. As the current regulations allow for faster progression, especially considering facilities planned in the near-term future, the FEA might be followed up in the long-term.

Regarding technical regulations, concepts for near-term and longer-term developments were identified. The near-term objectives are to establish a checklist for construction and operation license applications, and to create a guidance document specifying design-basis accidents and the required level of analysis . A key long-term objective is the development of a comprehensive technical regulatory framework for facilities with respect to the hazard potential that could potentially reach levels comparable to nuclear facilities.

In addition, potential proliferation issues connected to fusion facilities were analyzed, as well as challenges resulting from radioactive waste management.

Accident analysis is essential in fusion regulation. Based on a comprehensive literature survey on the available simulation codes and an analysis of the capabilities of the GRS code package AC^2, the development needs of AC^2 were identified.

Comparison of the radiological hazard potentials of high-activity sealed radioactive sources, nuclear facilities and fusion facilities

Dirk RADLOFF (Karlsruhe Institute of Technology (KIT))
Xue Zhou JIN (Karlsruhe Institute of Technology (KIT))
Joachim HERB (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH)
Clemens HEITSCH (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH)
Manuela JOPEN (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH)
Michael KOWALIK (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH)
Michael KUND (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH)
Ilka PETERMANN (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH)
Kevin COHRS (TÜV Rheinland)
Jens-Uwe SCHMOLLACK (TÜV Rheinland)
Matthias ENGLERT (Öko-Institut, Institute for Applied Ecology)
Anna KOPP (Öko-Institut, Institute for Applied Ecology)
Christoph PISTNER (Öko-Institut, Institute for Applied Ecology)
Philipp SAUTER (Max-Planck-Institute for Comparative Public Law and International Law)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

The objective of the pilot project ReFus were to develop concepts for the legal and technical regulation of fusion facilities in Germany. In order to develop a technical regulatory framework incorporating a graded approach, it is important to compare the radiological hazard potential of fusion facilities to other facilities regulated by either a radiation protection regulation framework or a nuclear regulation framework.

In a first step, the metric defined by the German Radiation Protection Ordinance, which is a multiple of the exemption limit of the different nuclides, was used. In a comprehensive literature review, the radioactive inventories of high-activity sealed radioactive sources (HASS) in Germany and different nuclear facilities like a nuclear power reactor, different research reactors, and a typical spent fuel cask were identified. For fusion facilities, the tritium inventory according to the licensing limits of ITER, the dust content inside the vacuum vessel and its nuclide composition for ITER and EU DEMO, as well as estimates of the activated corrosion products were used. The comparison was based solely on the radiological inventory and did not account for its releasability. The results showed that the unmitigated radiological hazard of fusion facilities is approximately at level with the largest facilities using HASS and reaching the range of research reactors.

In a second step, exemplary release calculations were performed for typical nuclides of fission and fusion facilities. These calculations considered the different half-lives of the nuclides and their behavior in the atmosphere. The results showed that among the fusion nuclides examined, tritium was the dominant nuclide influencing the dose assessment. The contribution of the other nuclides to the ingestion pathway was found to be insignificant due to their short half-lives compared to the contribution of tritium.

Assessment of Shutdown Dose Rates Following the Final Deuterium–Tritium Campaign at JET

Ylenia KOGOVŠEK ŽIBER (Institute Jožef Stefan)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

Accurate evaluation of shutdown dose rates is essential to ensure personnel safety during maintenance and decommissioning of fusion facilities and is a fundamental requirement in reactor design and operation. This study presents shutdown dose rate calculations for the JET tokamak after its final deuterium–tritium campaign and clean up phase prior to decommissioning. The primary objective is to evaluate the computational predictions by comparing them with experimental measurements.

The calculations were performed using the JSIR2S code developed at the Jožef Stefan Institute, which implements the rigorous two-step methodology. In this workflow, particle transport simulations with MCNP are coupled with activation and inventory calculations using FISPACT-II. The JSIR2S code has previously been validated against measurements from the TRIGA research reactor, and the present analysis extends this validation to fusion-relevant conditions.

The predicted shutdown dose rates agree well with measurements for predominantly deuterium–tritium (DT) plasmas, while larger deviations are observed for predominantly deuterium–deuterium (DD) plasmas. These discrepancies require further investigation. The results suggest that improved spatial discretisation, particularly near the ionisation chambers, could enhance accuracy. However, finer mesh resolution substantially increases computational cost, necessitating a balance between statistical uncertainty and calculation time.

To improve efficiency in photon transport simulations, weight windows derived from adjoint photon flux calculations with ADVANTG were tested. This variance-reduction approach does not include secondary photon sources, as they are not modelled in ADVANTG, while the photon source in the JSIR2S workflow is mesh-based and therefore not straightforward to treat consistently. The resulting weight windows reduced calculation time by up to 20% and improved uncertainty by up to 50%.

The calculated shutdown dose rates are compared with experimental data and with results obtained using other computational tools, enabling a comprehensive evaluation of the methodology under representative fusion operating conditions.

#13635

MELCOR and ASTEC codes benchmark study of ex-VV LOCA accidents impacting DEMO tokamak building.

Danilo Nicola DONGIOVANNI (ENEA NUC)
Mantas POVILAITIS (LEI)
Tommaso GLINGLER (Sapienza University of Rome, Department of Electrical and Energy Engineering (DIEE), Roma)
Matteo D'ONORIO (Sapienza University of Rome, Department of Electrical and Energy Engineering (DIEE), Roma)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

Exploitation of nuclear fusion as energy generation technology presents attractive features in terms of long-term sustainability justifying the needed efforts to overcome scientific and technological challenges in the path to a fusion power plant commercialization. Despite its intrinsic safety features derived by absence of runaway processes, fusion plants still present some mobilizable radiological inventories hazard whose associated risks of release shall be minimized. Therefore, European DEMOnstration plant conceptual design includes deterministic safety assessment of events potentially leading to radiological releases. Tokamak building together with its atmosphere processing systems are plant systems relevant to safety since implementing a radiological inventory confinement function. Ex Vacuum Vessel (VV) Loss of Coolant Events (LOCA) events are among studied events since helping gaining insight in the building capability to mitigate radiological releases and associated safety consequences.
The assessment of such consequences relies on safety codes able to model plant systems and phenomenology involved in the accidental transient to deterministically quantify accident consequences in terms of releases. MELCOR and ASTEC (Accident Source Term Evaluation Code) are two integral codes able to perform severe accident analyses in nuclear fission (LWR, PWR) and adapted to address relevant phenomena occurring in nuclear fusion plant accidents.
In present analysis a model of the DEMO Tokamak building including active and passive confinement features is developed and implemented both in MELCOR and ASTEC. A set of ex vessel LOCA events were studied involving systems with different enthalpic content, operating at different thermodynamic points (T,P) and having different locations/elevations in plant. In particular Divertor Cassette / Plasma Facing Unit (PFU) HTS loops and Water Cooled Lithium Lead (WCLL) Breeding Blanket / First Wall (FW) heat transfer system (HTS) exVV LOCA events were considered. Since assuming multiple independent cooling loops failures as initiators, such events fall in the Design Extension Conditions category. The evolution of accidents has been studied with both codes and results agreement in terms of main observables (rooms pressurization, atmosphere and walls temperature, mobilized inventories) is assessed and discussed.

ENHANCING TOKAMAK OPERATION TRAINING THROUGH MULTI-ROLE INTERACTIVE SIMULATION USING SOPHIA

Filip JANKY (Tokamak energy ltd.)
Ian ROBERTS (Tokamak energy ltd.)
Alexei DNESTROVSKII (Tokamak energy ltd.)
Sergei MEDVEDEV (Tokamak energy ltd.)
Paul SMITH (School of Computing and Communications, Lancaster University)
Colin BOXALL (School of Engineering, Lancaster University, Lancaster)
Samuel MURPHY (School of Engineering, Lancaster University, Lancaster)
Jari VARJE (Tokamak Energy Ltd.)
Hanni LUX (Tokamak Energy Ltd.)
Maria MICHALOPOULOU (Tokamak energy ltd.)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

The SOPHIA tokamak simulator, originally developed for validating plasma scenarios and verifying discharge compliance within operational limits, has been evolved into a pedagogical platform to enable students to deepen their understanding of plasma physics and reactor control through hands-on interaction in an immersive fusion control room environment. This extension transitions SOPHIA from a purely predictive tool to an active, multi-operator learning environment that simulates real-time collaborative operations. The architecture introduces discrete functional roles — Vacuum Operator, Glow Discharge Operator, Power Supply Operator, Neutral Beam Operator, and Pilot — necessitating rigorous system preparation and inter-operator communication prior to pulse execution. These roles were selected to show how key aspects of tokamak design exist to ensure optimum plasma conditions enhancing the learning experience. For example, the Vacuum Operator and the Glow Discharge Operator roles link to creating the vessel condition necessary for plasma breakdown and ensuring the plasma remains free from impurities, the Neutral Beam Operator role links to plasma heating, the Power Supply role links to equilibrium reconstruction and all roles contribute to delivering a desired plasma scenario. The simulation engine enforces operational constraints by providing real-time feedback and inhibiting discharges in response to incorrect subsystem configurations. As part of the LUNAR project, this educational iteration of SOPHIA will be deployed at Lancaster University to facilitate the study of complex, interactive tokamak operations. This paper details the software architecture of these new roles and discusses how this immersive, team-based approach enhances student understanding of the complexities inherent in fusion device operations.

Activation of Fusion Reactor Blanket Materials and Their Role in Radioactive Waste Management

Asset SHAIMERDENOV (The Institute of Nuclear Physics)
Darkhan SAIRANBAYEV (The Institute of Nuclear Physics)
Timur KULSARTOV (The Institute of Nuclear Physics)
Magzhan AITKULOV (The Institute of Nuclear Physics)
Saulet ASKERBEKOV (The Institute of Nuclear Physics)
Batyr SHAKIROV (The Institute of Nuclear Physics)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

A fusion reactor is a technically and technologically complex facility in which nuclear fusion reactions of light hydrogen nuclei occur, resulting in the release of a significant amount of energy and radiation (neutrons, neutrinos, and gamma radiation). All in-vessel components, including blanket materials, are exposed to intense radiation fields – predominantly neutron irradiation – which leads to degradation of their physical and mechanical properties as well as neutron activation, particularly of impurity elements present in the initial material composition. During the subsequent decommissioning of fusion reactor components, their induced radioactivity must be carefully taken into account in order to develop an appropriate radioactive waste management strategy.

In this work, a comparison is presented between experimental and calculated data on the activation of fusion reactor blanket materials that were irradiated in the WWR-K research reactor under a thermal neutron spectrum. The irradiation conditions relevant to the activation of radioisotopes in the materials are analyzed. In addition, modeling results are presented for predicting the activation of blanket materials under irradiation by a fusion neutron spectrum (14 MeV). The selectivity of radioisotope activation is demonstrated, and recommendations for the selection of materials suitable for fusion reactor blanket applications are formulated.

#13731

High TRL Tritium Handling Subsystems for Demonstration Fusion Machines

Walter SHMAYDA (Tritium Solutions, Inc.)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

Fusion power development has transitioned from national laboratories to the commercial sector. A plethora of fusion startups are proposing a variety of pathways to fusion power by tapping into inertial, magneto-inertial and magnetic confinement schemes. In almost every case, the funding plan relies on demonstrating the viability of the approach to investors with a demonstration device before the investors commit to the construction of a fusion power plant. Most of these approaches will rely on the DT fuel cycle. As these startups transition from deuterium/deuterium operations to deuterium/tritium operations, they encounter a bewildering number of tritium related decisions that can strain their financial portfolio. Research efforts and funding should focus on fusion machine fueling, effluent recovery and tritium extraction from breeder blankets rather than on the primary fuel cycle tritium handling subsystems.

This presentation will demonstrate that may of the fuel cycle tritium subsystems are concept agnostic and that the technology readiness level of these common tritium handling subsystems is in the 8 to 9 range for demonstration fusion machines. These engineered subsystems have extensive tritium operational experience, have demonstrated negligible environmental impact and only require minor throughput adjustments to accommodate specific fusion machine throughput requirements. These subsystems include Tritium Storage and Delivery, Isotope Separation, Emission Mitigation System (EMS), Water Treatment, and Secondary Enclosure Cleanup (SEC) for gloveboxes that use inert cover gases. For example, selecting the appropriate Isotope Separation System depends on the application: the Thermal Cycling Absorption Process (TCAP) for protium stripping at low throughputs or the low-inventory wet-wall cryogenic column for deuterium/tritium rebalancing (or separation) at higher throughputs. The Emission Mitigation System, a third-generation design, relies on a classical burn and dry approach followed by tritium concentration in water using the Catalytic Exchange Combined Electrolysis (CECE) process before a slip stream of electrolyzed DT is directed to the Isotope Separator for tritium reclamation and re-use. The combination of the EMS and the CECE subsystems will be critical in minimizing the tritium environmental footprint of fusion machines. SEC provides a defense in depth strategy that not only captures any accidental tritium losses from the Tritium Storage and Delivery but provides a pathway for the return of escaped tritium to the fuel cycle.

CFD simulation of a LOVA in DEMO tokamak with D/T-air reactions

Jean-François CIPARISSE (ENEA)
Danilo Nicola DONGIOVANNI (Centro Ricerche ENEA Frascati,Via Enrico Fermi 45, 00044 Italy)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

In the framework of safety issues assessment in nuclear fusion, one of the main concerns is a fire of the D/T mixture contained in the vacuum vessel (VV) following a Loss Of Vacuum Accident (LOVA). This event leads to the formation of a reactive environment (D/T-air mixture) and could cause a fire and a leak of tritiated water vapor, equipment damage and. In the worst scenario, ignition of activated dust and its subsequent release in the Tokamak building could occur as well.
Previous numerical studies on a 20-L sphere have been carried out to assess the risk of a hydrogen/air reaction in two different scenarios. In the first one, an ingress of ambient air in a subatmospheric vessel containing an equimolar mixture of D/T led to an ignition by compression of the reactants. In the second one, the same vessel was filled with a lean D/T-air mixture at atmospheric pressure and part of the walls was heated. In this case, the mixture ignited as well.
Moving from both studies, the combination of LOVA and hot walls was considered as a realistic case to reproduce numerically in a full-scale Tokamak. In this paper, as exemplificatory scenario, the break of a EWG line of ECH system in DEMO plant was chosen as initiator of LOVA accident. A model was developed with Melcor code to have an estimate of initial air mass flow at launcher within VV. Then, a CAD model of DEMO VV full size was implemented in CFD code OpenFOAM.
As initial conditions, an equimolar mixture of D/T was considered to fill the VV. The mass of the mixture is 4.3 kg and all walls are hot. A 21-reaction detailed hydrogen-air mechanism was adopted to take into account chemistry. The gas mixing transient is simulated during 5 s and the possibility of an ignition of the D/T-air mixture is investigated.

Optimization of the HRNS beam dump for reduced activation and radiological impact

Anna WÓJCIK-GARGULA (Institute of Nuclear Physics Polish Academy of Sciences, Radzikowskiego 152, PL-31342 Krakow, Poland)
Urszula WIĄCEK (Institute of Nuclear Physics Polish Academy of Sciences, Radzikowskiego 152, PL-31342 Krakow, Poland)
Mehrara ROJIN (Institute of Nuclear Physics Polish Academy of Sciences, Radzikowskiego 152, PL-31342 Krakow, Poland)
Grzegorz TRACZ (Institute of Nuclear Physics Polish Academy of Sciences, Radzikowskiego 152, PL-31342 Krakow, Poland)
Giovanni MARIANO (ITER Organization, Route de Vinon-sur-Verdon, CS 90 046, 13067 St. Paul Lez Durance Cedex, France)
Bruno CORITON (ITER Organization, Route de Vinon-sur-Verdon, CS 90 046, 13067 St. Paul Lez Durance Cedex, France)
Ryszard KANTOR (Cracow University of Technology, ul. Warszawska 24, 31-155 Cracow, Poland)
Jarich KONING (Heemskerk Innovative Technology, Rotterdamseweg 183C, 2629 HD, Delft, The Netherlands)
Dariusz MORAWSKI (Institute of Nuclear Physics Polish Academy of Sciences, Radzikowskiego 152, PL-31342 Krakow, Poland)
Marek SCHOLZ (Institute of Nuclear Physics Polish Academy of Sciences, Radzikowskiego 152, PL-31342 Krakow, Poland)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

The High Resolution Neutron Spectrometer (HRNS) requires a beam dump ensuring reliable operation while minimising radiological hazards to personnel and equipment. A systematic activation analysis was performed for boron carbide (B_4C), stainless steel (SS316LN-IG), and tungsten, considering their short- and long-term behaviour in the HRNS geometry. Monte Carlo neutron transport coupled with activation calculations was used to determine total activity, dominant radionuclides, decay trends, and shutdown dose rates (SDR).

All materials exhibit high initial activity dominated by short-lived products. Tungsten accounts for approximately 98% of the total radionuclide inventory at 1 day after shutdown due to W 185 and W 181, resulting in the highest early SDR; activity decreases rapidly within the first months. Boron carbide is dominated in the long term by tritium produced through neutron reactions on boron isotopes. Although it contributes ~97% of the total activity after 10 years, it generates negligible external gamma radiation. Stainless steel shows lower initial activity, but Co-60 governs the long-term gamma dose. The residual inventory is dominated by Fe-55 and Ni-63, relevant for waste classification rather than external dose.

Based on these results, several optimisation strategies were assessed. Partial hollowing of the tungsten core reduces mass and peak short-term activity while maintaining neutron absorption. Variable B_4C thickness—greater where tungsten is removed and reduced where it remains solid—ensures uniform moderation and protection of the SS316LN-IG shell. The use of reduced-activation or low-cobalt alloys for structural components is identified as an effective measure to lower long-term gamma dose. Layered configurations combining tungsten, B_4C, and stainless steel enable optimisation of mechanical strength, neutron attenuation, and activation performance.

This analysis demonstrates that appropriate material selection and geometry optimisation significantly reduce radiological impact and improve maintainability of fusion neutron instrumentation. Coupled transport and activation simulations provide a robust basis for radiological optimisation of high-flux beam dump systems.

Acknowledgements: This work was done as part of a project co-financed by ITER Organization, France and the Polish Ministry of Science and Higher Education under the program entitled “International Projects Co-financed". We gratefully acknowledge the Polish high-performance computing infrastructure PLGrid (HPC Center: ACK Cyfronet AGH) for providing computer facilities and support within computational grant no. PLG/2025/018826. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

AI?Driven Engineering Knowledge Base and Agentic Workflow System for Accelerating Technical Activities at ITER

Junmin YANG (ITER Organization)
Jean-Daniel DELAPLAGNE (ITER Organization (IO))
Maria ORTIZ DE ZÚÑIGA (Fusion for Energy (F4E))

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

ITER's complex engineering environment produces a vast and continuously expanding body of structured and unstructured information, including CAD data, analysis reports, system design etc.. Efficient access to this knowledge—and the ability to reason across multiple heterogeneous sources—is essential for high quality engineering, timely design reviews, and informed technical decision making. Traditional search mechanisms often remain limited to keyword lookup and rigid metadata filters, offering little support for contextual interpretation, cross document reasoning, or engineering specific workflows.

To address these challenges, our solution aims to index, structure, and interconnect engineering datasets document contents into a unified semantic knowledge layer. Leveraging modern vector search technologies and large language models, the system enables users to query the knowledge base in natural language, retrieve relevant information across multiple data sources, and obtain synthesized outputs instead of long document lists.

Beyond search, the solution introduces an agentic workflow engine capable of performing multi-step reasoning over documents. Agents can decompose complex engineering questions, consult domain-specific knowledge, analyze relevant technical documents, and generate concise, traceable answers. These capabilities can support daily engineering tasks such as design review. By automating the information gathering and preliminary reasoning phases, the system reduces manual effort and enables engineers to focus on expert judgement rather than document hunting.

This project advances ITER's efforts in AI automation. By combining semantic indexing, multi-source knowledge integration, and agentic reasoning workflows, it offers an intelligent platform to improves knowledge discoverability and enhance productivity, supporting higher quality engineering decisions across the organization.

#13854

Nuclear safety and radwaste aspects of the IFMIF-DONES lithium target assembly

Arkady SERIKOV (Karlsruhe Institute of Technology (KIT))
Sergej GORDEEV (Karlsruhe Institute of Technology (KIT))
Yuefeng QIU (Karlsruhe Institute of Technology (KIT))
Dieter LEICHTLE (Karlsruhe Institute of Technology (KIT))
Concepcion OLIVER (CIEMAT, National Fusion Laboratory, Avenida Complutense 40, 28040 Madrid)
Danyl PÉREZ SÁNCHEZ (CIEMAT, National Fusion Laboratory, Avenida Complutense 40, 28040 Madrid; Consorcio IFMIF-DONES España, Granada)
Jorge MAESTRE (Consorcio IFMIF-DONES España, Granada)
Francisco MARTIN-FUERTES (CIEMAT, National Fusion Laboratory, Avenida Complutense 40, 28040 Madrid; Consorcio IFMIF-DONES España, Granada)
Claudio TORREGROSA (Consorcio IFMIF-DONES España, Granada)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

The IFMIF-DONES construction progress requires comprehensive nuclear safety and radiation waste management assessments of the components exposed to neutron irradiation. The neutron source is formed on the deuteron-lithium (d-Li) stripping nuclear reactions by impinging the 40 MeV accelerated deuteron beam with 125 mA current on the high-speed circulated liquid Li target, producing a total neutron flux of 1.2e15 n/cm2/s, with the maximum deuteron heat deposition of 110 kW/cm3 at the Bragg peak surface inside the Li target layer at the 20x5 cm2 deuteron footprint area. The liquid Li speed of 15 m/s provides effective cooling of the target assembly and removes the high-temperature Li to the quench tank. This neutronics work demonstrates that the safe functionality of the large IFMIF-DONES facility depends on the safe operation of the Target Assembly (TA), where the d-beam avoids interactions with uncooled structural components, such as the Li-inlet and outlet channel, and backplate. Overall, the nuclear safety analysis of the Target Assembly (TA) was performed using a series of coupled simulations, including deuteron beam dynamics, neutronics, thermal-hydraulics, and the thermo-mechanical modeling of the TA components. The precise neutronics modeling is achieved using the MCNP6.2 radiation transport code, CAD-based geometry, and defining an updated 1 mm fine-resolution d-beam profile at 9 cm upstream in front of the Li-target surface to detect any possible interactions of the d-beam with the TA structures. As the input data for the MCNP6.2 calculations, the d-beam profile has been produced in the beam dynamic analyses, taking into account the static and dynamic beam errors, as well as thermal expansion of the Li-target. Two safety-related d-beam profiles have been studied: the reference and the worst-case 9-cm upstream beam profiles. The definition of the worst-case beam profile encompasses all possible misalignments with respect to the Li target, including an additional 3 mm shift of the beam in the horizontal direction, where the d-beam periphery touches the Li-channel side wall.

To support the IFMIF-DONES licensing and TA radwaste assessments and management, activation analyses were performed for the backplate, the most radioactive component of the Target Assembly. The study evaluated two candidate materials, SS316Ti and Eurofer-97, using the MCNP6.2 and FISPACT-II codes. Specific (Bq/kg) and total (Bq) activities were determined for all identified radioisotopes to ensure compliance with Low- and Intermediate-Level Radioactive Waste (LILW) acceptance criteria for disposal by ENRESA at the El Cabril facility in Spain.

VALOFUSION: Coupling of a Cold Crucible Induction Furnace with a Catalytic Membrane Reactor for the Detritiation of Metallic Waste

Aurélien CHASSERY (CEA)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

Currently under construction at the Cadarache site in the south of France, the ITER reactor aims to generate energy by reproducing the thermonuclear fusion reactions that occur in the Sun. Once operational, the reactor is expected to produce tritiated solid waste, meaning waste that contains tritium.

Tritium is a radioactive element that is highly mobile in the environment, and its migration is difficult to control. This makes the management of tritiated waste particularly challenging during both interim storage and final disposal. One possible approach is to treat such waste using a decontamination process designed to remove the tritium (detritiation) before the waste is sent for storage and disposal. The VALOFUSION project focuses specifically on purely tritiated metallic waste (i.e., waste that does not contain any other radioactive elements), for which existing treatment technologies remain limited.

The VALOFUSION project aims to:

• develop a decontamination process based on the melting of tritiated metal in order to extract the tritium and homogenize the remaining contamination within the molten metal;

• enable the recovery of the extracted tritium so that it can be reused as fuel in the ITER reactor. To achieve this, the melting furnace will be coupled with a catalytic membrane reactor. This reactor converts the tritiated water produced during the melting of tritiated metallic waste into tritiated hydrogen, which can then be selectively extracted through permeation.

An optimization of the operating parameters made it possible to validate the detritiation step using the induction furnace, achieving detritiation factors greater than 1300. Coupling tests between the induction furnace and the catalytic membrane reactor confirmed the proper operation of the overall process.

Tritium Accumulation in Egeria Densa Growing in Tritiated Water

Shogo SAIKI (Kyushu University)
Kazunari KATAYAMA (Faculty of Engineering Sciences, Kyushu University)
Kenta AKASHI (Faculty of Engineering, Kyushu University)
Mawufemor ZIGAH (Interdisciplinary Graduate School of Engineering Sciences)
Hikari INOUE (Interdisciplinary Graduate School of Engineering Sciences, Kyushu University)
Michael PORTUPHY (Faculty of Engineering Sciences, Kyushu University)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

Nuclear fusion reactors using deuterium and tritium fuel are considered a promising next-generation energy source. Since tritium is a radioactive material, understanding its environmental behavior in the event of an accidental release is crucial for safety assessments and public acceptance of the fusion reactors. A part of tritium released into the environment enters aquatic ecosystems, mainly as tritiated water, and is taken up by aquatic plants. Tritium incorporated into aquatic plants may be transferred to humans via the food chain, involving fish and other organisms within these diverse ecosystems. Therefore, understanding the environmental behavior of tritium is of great importance for accurate risk assessment. Most of the tritium taken up by plants exists as tissue free water tritium (TFWT), but some of tritium is bound to tissues through growth processes such as photosynthesis and exists as organically bound tritium (OBT). In particular, non-exchangeable OBT (NE-OBT) is of concern because of its long biological residence time. In this study, aquatic plants were cultivated in tritiated water at different concentrations to investigate the relationship between the cultivation period and the amount of tritium incorporated into the plants.

The tritium concentration in the cultivation water was set at 10 kBq/mL for Group A and 1 kBq/mL for Group B. Three shoots of Egeria densa were cultivated in each group. Starting from the onset of cultivation, one shoot was collected from each group every three days. After surface moisture was carefully removed, the weight of each sample was measured. The samples were then placed into quartz tubes for tritium speciation analysis. TFWT was extracted by vacuum drying, while exchangeable organically bound tritium (E-OBT) and non-exchangeable OBT (NE-OBT) were recovered by air drying and thermal combustion, respectively. The tritium collected in each water bubbler was subsequently measured using a liquid scintillation counter.

The TFWT of Group A was approximately 10 times higher than that of Group B, suggesting that the TFWT is proportional to the tritiated water concentration. The fraction of NE-OBT was only a few percent of the total tritium recovered from the plant samples. The amount of NE-OBT increased with the duration of cultivation, confirming its formation through photosynthesis processes. However, no significant difference in the total amount of NE-OBT was observed between Group A and Group B, suggesting that tritium concentration in the cultivation water had no measurable effect on NE-OBT formation.

#13977

Evaluation of tritium mass transfer rates from liquid lithium to the gas phase and yttrium

Akira TERUYA (Kyushu University)
Yuuka MORI (Kyushu University)
Kazunari KATAYAMA (Kyushu University)
Kentaro MASUTA (Kyushu University)
Hiroki ISOGAWA (Kyushu University)
Akashi KENTA (Kyushu University)
Makoto OYAIZU (National Institute for Quantum Science and Technology (QST))
Yuto IINUMA (Institute for Integrated Radiation and Nuclear Science, Kyoto University)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

Construction of A-FNS is planned in Japan as a material irradiation facility to evaluate material damage by neutrons from a nuclear fusion reaction. The reaction of flowing lithium (Li) with deuteron beams to produce high-intensity neutron beams generates tritium as a by-product. For safety operation of the facility, it is important to understand tritium behavior in Li and to control tritium appropriately. The recovery of hydrogen isotopes from liquid Li is expected to be achieved using an yttrium (Y) hot trap. In our previous study optimizing the hot trap designs, we attempted to determine a mass transfer coefficient representing tritium absorption rate from liquid Li into Y by heating neutron-irradiated Li samples. We observed that significant amount of tritium was released from liquid Li into the gas phase, which could not be disregarded for estimating tritium absorption rate into Y. Therefore, in this study, we attempted to quantify a mass transfer coefficient representing the release rate of tritium from liquid Li to the gas phase with and without Y chips.

The Li sample irradiated by neutron at the Kyoto University Research Reactor was set in a molybdenum crucible with or without Y chips, and the crucible was placed in a quartz tube. The crucible was heated to 400℃ while flowing argon gas, and the released tritium was collected in water bubblers. The water bubblers were replaced at 10-minute intervals to monitor the time evolution of the tritium release rate into the gas phase. The rates of tritium transfer to the gas phase and absorption into Y are assumed to be proportional to the tritium concentration in the liquid Li, which is considered uniform throughout the phase. Based on these assumptions, mass balance equations were formulated to analyze the temporal evolution of the tritium release rate. The mass transfer coefficient, k_g, for tritium release from the liquid Li without Y was determined to be 3.1×10-6 m/s by fitting calculated release profiles to the experimental data. In contrast, the coefficient in the presence of Y, k_gY, was found to be 2.0×10-7 m/s, indicating a significant reduction in the release rate. The cause of this difference is currently under investigation. In addition, the mass transfer coefficient of tritium absorption rate into Y, k_LiY, was estimated to be 6.1×10-8 m/s. Additional measurements at various temperatures and supplementary experiments using deuterium gas are currently underway; the latest results will be presented at the conference.

Key enablers of fusion deployment at scale

Chiara BUSTREO (Consiglio Nazionale delle Ricerche)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Oral

Abstract

Fusion is a promising sustainable energy source that can potentially contribute to achieving the most ambitious climate goals with firm carbon-free baseload electricity. Due to recent advancements, fusion is transitioning from a purely research stage to a pre-commercialisation phase. However, many aspects related to its commercialisation are uncertain. Energy scenarios within the EUROfusion TIMES Model have been used to explore the conditions that could enable fusion to play a significant role in a future decarbonised global energy system, assuming moderate socio-economic development. The results indicate that a maximum share of 21% in a ~140 PWh global electricity generation is achievable by 2100 under the following concurrent conditions: fusion is commercialised by 2040; the capacity doubling time is as large as that experienced by fission in the ‘70s and a stable tritium supply chain is established; the average overnight cost of the first fleet of fusion plants does not exceed 7200 USD/kW with a learning rate of 10%, and solar and wind generation is limited to 75% in each model timeslice. In contrast, the share of fusion under the most conservative assumptions reaches only a few percent, despite corresponding to an operating fleet comparable in size to the current fission fleet. The study also proves two insights: firstly, that early availability alone cannot guarantee high fusion shares in power markets. Secondly, deployment speed along with restriction to variable renewable energy (VRE) contribution are dominant structural factors. The presentation will also discuss the role of the same factors under a more conservative socio-economic development, reflecting a fragmented world derived from the current uncertain geopolitical context.

Qualification of ITER Equatorial Launcher Feedthroughs: Implementing an "ISO 19443 Plus" Approach for First Confinement Components

Vincenzo MARCELLETTI (BERTIN)
Guillaume REVILLOD (Bertin)

Topics

K. Safety and Environment, Socio-economic studies and Technology Transfer

Communication type

Poster

Abstract

ITER's Equatorial Launcher (EL) diagnostics require electrical and optical feedthroughs capable of maintaining first confinement barrier integrity under extreme operational and accidental conditions. This poster discusses the qualification process of commercial-off-the-shelf (COTS) feedthroughs integrated into confinement feedthrough (CF) prototypes for PBS 52.EL, demonstrating how Bertin Technologies' enhanced quality methodology ensures compliance with stringent nuclear safety requirements while optimizing cost and schedule.

The qualification campaign addresses multiple challenge layers: functional performance verification (leak-tightness $\leq 1\times10^{-10}$ Pa·m^3/s, optical insertion loss, insulation resistance), environmental testing (thermal cycles up to 250 °C, outgassing rates), and accidental scenarios (abnormal pressure, seismic/inertial loads, fire resistance). To manage this complexity for first confinement components, Bertin has developed an "ISO 19443 Plus" methodology that strategically combines ISO 19443:2018 baseline requirements with key elements from ASME NQA-1 (QA records rigor, personnel qualification) and 10 CFR regulatory principles (formal Nuclear Safety Control Plans, DR-level traceability matrices). This approach enables risk-based optimization of qualification strategies—such as similarity analyses for reference selection and targeted test campaigns—while maintaining full regulatory compliance and traceability.

The Nuclear Safety Officer (NSO) coordinates the end-to-end process: from Defined Requirements (DR) identification through Technical Requirements (TR) propagation to suppliers, Manufacturing & Inspection Plans (MIP), multi-stage Inspection & Test Plans (ITP) covering functional/environmental/accidental tests, and final Quality Compliance Records (QCR) demonstrating conformity. Special process qualification (welding of COTS assemblies), supplier control for non-standard optical fibers, and management of Hold/Witness points with ITER ensure robust execution.

The poster presents the EL feedthrough qualification framework, practical interfaces between NSO, engineering and QA teams, test results synthesis, and lessons learned showing how this methodology delivers superior traceability, regulatory compatibility, and demonstrable efficiency for safety-critical fusion reactor components operating under the most demanding conditions.