Overview of CEA neutronics activities in fusion reactors

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

The investigation of energy production is one of the main research areas of the French Alternative Energies and Atomic Energy Commission (CEA). Although the largest share of R&D efforts is devoted to fission energy, due to the contribution of this source to the electricity mix of France (57 reactors currently operated by EDF), fusion energy is also strongly represented. In recent years, resources devoted to fusion at CEA have been consistently expanding. Fusion neutronics activities at CEA concern a large set of fusion facilities in Europe. The operative facility is WEST, a plasma physics and first wall technology experimental facility at IRFM (Institute for Magnetic Fusion Research), but CEA is also involved in the JET (UK) facility analyses, as well as in the international ITER and DEMO reactors. Recently, CEA also started collaborating with the CHARTREUSE reactor design from the French Renaissance Fusion start-up within a collaboration supported by the French BPI. The activities carried out in this context involve different topics like neutronics code development, new methodologies implementations, Monte Carlo code and nuclear data validation and nuclear analyses. In WEST, CEA is involved in the calculation of waste inventories and the investigation of practical issues like the ones due to runaway electron activation. Regarding JET, CEA is involved in streaming (STRE) and shutdown dose rate (SDR) simulations of the fusion facility within the EUROfusion PrIO work package. Various experimental campaigns are being investigated using numerical simulation tools. Specifically, the Deuterium–Tritium (DTE2-DTE3) JET campaigns are under analysis with specific computational tools coupling radiation transport and material activation within TRIPOLI-4® Monte-Carlo code. Among these numerical tools, it is worth mentioning the Rigorous-Two-Step method (R2S) for activation calculations using Monte Carlo codes. The implementation of the Direct-One-Step (D1S) method will be also considered in near future. CEA teams are also involved in the assessment of ITER neutronics by contributing to the design of specific diagnostics with F4E, by participating to the ACP neutronics benchmark, and by analyzing the possible impact of runaway electron activation in the recent re-baselining of the reactor and its possible impact on the assembly program of the DT-1 phase. Regarding the DEMO prototype reactor, CEA investigates the breeding blanket concepts and related structures with TRIPOLI-4®. In particular, the nuclear analysts have focused on the assessment of the neutronics performance of the WCLL BB within the WPBB EUROfusion package. The main goals of these investigations concern the assessment of the tritium-breeding ratio (TBR) to ensure tritium self-sufficiency, as well as indirect damages on materials (DPA, helium production, and energy deposited in the first wall). CEA develops its own TRIPOLI Monte Carlo codes and implements most of the required functionalities in Fusion Neutronics, like D1S and R2S calculation schemes, or specific neutron transport libraries. The TRIPOLI-4® code simulates the coupled transport of neutron and photons, and is thus used to simulate most of the typical problems for WEST, ITER, CHARTREUSE or DEMO. It is widely used for validation purposes in fusion benchmarks in FNG at ENEA, FNS at JAERI, or OKTAVIAN at Osaka University. Moreover, TRIPOLI-4® natively enables the simulation of the electromagnetic shower, which allows addressing charged particle transport and the related challenges. Regarding innovative topics, CEA develops a roadmap in the field of Virtual Reality and online dose rate calculation. CEA also recently carried out several first of a kind experiments on electronics, notably under deuterium fusion neutrons at its WEST tokamak facility in Cadarache and under deuterium-tritium fusion neutrons at the UKAEA's JET facility in Culham (UK), to assess the sensitivity of electronics to neutron-induced single event effects (SEEs) in the neutron environment of a fusion facility and to compare it with the sensitivity in the natural terrestrial atmospheric neutron environment for which electronic components are typically qualified by manufacturers. The models and methods validated by these experiments will now be used by the NERITA (Neutron Electronics Reliability In Tokamaks and Accelerators) project led by CEA and CERN with other academic partners, aiming to replace the qualification of electronic systems to SEEs induced by the tokamak's neutron environment (which is not feasible for commercial-of-the-shelf electronic systems due to the absence of traceability of the origin of their constituent semiconductor components) by the adaptation and qualification of the tokamak's neutron environment with respect to the SEEs it induces on general electronics. The analysis of these experiments included dedicated particle transport simulations with GEANT-4. The purpose of the proposed presentation is to illustrate the current activities at CEA and the related R&D program for the forthcoming 2026-2027 years.

Shutdown dose rate evaluation for the beam dump in the Linear IFMIF Prototype Accelerator (LIPAc) with Direct 1 Step MCNP

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	Oral Contribution	☐

As part of the IFMIF/EVEDA project, the Linear IFMIF Prototype Accelerator (LIPAc) has been constructed at Rokkasho in Japan. As shown on the left in Fig. 1, deuteron beam accelerated up to 5 MeV or 9 MeV is introduced into the 1.125 MW high-power beam dump delivered by Europe (CIEMAT). The beam dump is activated by nuclear reactions with deuterons and neutrons, which are generated secondary due to deuteron interaction with the copper cone in the beam dump. The shutdown dose rate (SDDR) due to the secondary neutrons has been calculated with Direct 1 Step (D1S) method for the beam dump. In addition, the SDDR calculations include the contribution from the activation due to nuclear reactions between the deuterons and the copper cone, Cu(d,x), by calculating the total amount of produced photon with FISPACT-II.

The right side of the Fig. 1 shows an example for the calculation result on the SDDR in the case of an operation time for 10^4 seconds, and a cooling time for 10^4 seconds after the operation. The presentation will report on the necessity of the SDDR evaluation considering the contribution of both the deuterons and the secondary neutrons since their contributions are different depending on the location, the deuteron energy, operation and cooling times. This present study also indicates that the operation scenario needs to be determined carefully for the hands-on maintenance because decay of the SDDR tends to be slow as operation times are prolonged. Additionally, in the case of 9 MeV deuteron beam operation, the SDDR due to Cu(d,x) reactions increases remarkably by approx. 350 times in comparison with 5 MeV operation because of the ^{65}Zn production when cooling times are among 10^6 – 10^8 seconds.

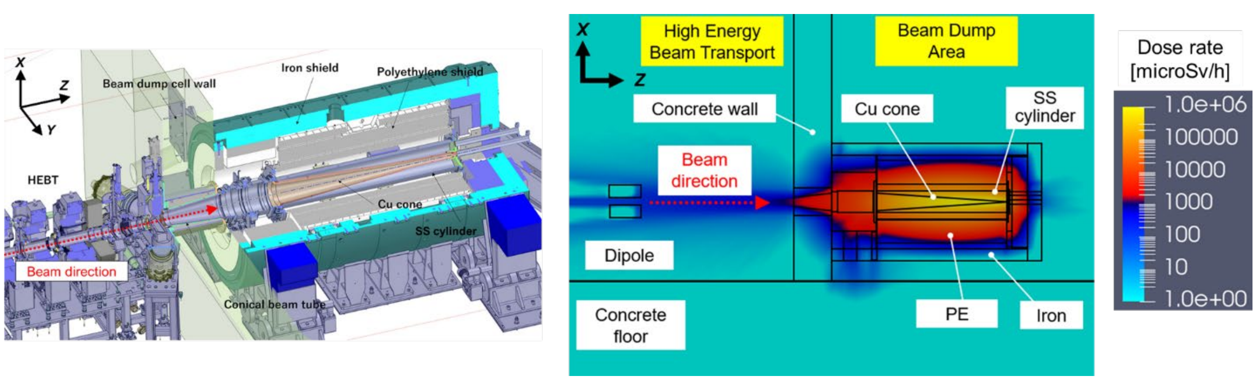

Fig. 1: (left) LIPAc beam dump, (right) Shutdown dose rate at Beam time = 10^4 seconds and Cooling time = 10^4 seconds for 5 MeV D^+ beam operation.

ITER Equatorial Port 9 Neutronics Analysis

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Please, indicate type of contribution:	Poster session ☐
	Oral Contribution ☒

Upon its completion, the ITER fusion device will be the world's largest tokamak, a toroidal machine designed to contain plasma through the use of strong magnets. As the largest fusion device ever to be constructed, ITER presents several unique challenges in terms of modeling, manufacturing, and transportation, to name a few. The size of the ITER machine necessitates the use of large and complex neutronics models and sophisticated tools to simulate the behavior of radiation inside the machine and the impact to equipment and structures nearby.

Access to the ITER machine for diagnostic, fueling, vacuum, tritium breeding etc. is provided via the lower, equatorial, and upper ports. ITER is designed with 18 equatorial ports, each with a specific set of equipment. The subject of this discussion is the neutronics analysis of equatorial port location 9 (EP9) under plasma operation (mode-0) and 10^6 s after shutdown during maintenance (mode-1). The US domestic agency, US-ITER, is responsible for integrating EP9 amongst other systems. EP9 is a so-called diagnostic port hosting three diagnostic systems, one in each of the Diagnostic Shield Modules (DSM). The Toroidal Interferometer and Polarimeter (TIP) system is used to measure the plasma density; the Electron Cyclotron Emission (ECE) system is used to provide high spatial and temporal resolution measurements of the electron temperature evolution and the electron thermal transport inferences; and finally, the visible/infrared Wide-Angle Viewing System (WAVS) will be used to provide visible and infrared viewing and temperature data of the first wall for its protection in support of machine operation. The systems modeled in the calculation for the port plug and interspace regions of this port are continuous through the bioshield plug and into the port cell.

The quantities calculated for EP9 during mode 0 are neutron and photon flux, nuclear heating, effective neutron and gamma dose, tritium production in B4C, helium production in B4C, dose to silicon at end-of-life (EOL), dose to polyethylene at EOL, and damage in terms of DPA in SS316LN-IG at EOL. In addition to these quantities, the shutdown dose rate, from mode-1 at 10^6 s after shutdown, was also considered.

Fusion Neutronics Developments for Fusion REactor Design and Assessment (FREDA)

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	Oral Contribution	☒

As part of the U.S. Department of Energy (DOE) Scientific Discovery through Advanced Computing (SciDAC) project, the Fusion REactor Design and Assessment (FREDA) team developed additional capabilities for fusion neutronics. The project focuses on three main aspects: 1) uncertainty quantification of modeling and simulation, 2) coupling with plasma physics and thermal hydraulics, and 3) expansion and validation of OpenMC.

For uncertainty quantification, the team implemented Total Monte Carlo (TMC) for nuclear data uncertainty propagation. The team is also exploring the uncertainty range of homogenization of materials in a geometry and how that uncertainty can be reduced by implementing scaling factors leveraging cross section data. Of additional interest is the impact of explicit operational modeling of pulsed operation on material activation and thermal hydraulics.

For coupling with other physics codes, the team is implementing fine-spatial-resolution coupling from OpenFOAM back to OpenMC. This is accomplished by automating the generation of new Cubit geometries whose subdivisions are based on the gradient of results (i.e. temperature, density) from OpenFOAM, which ensures a more accurate radiation transport calculation. An additional capability which was added is the ability to update the neutron source distributions in position and direction in OpenMC. This capability has been applied in a study comparing different fusion neutron direction biases from fusion fuel spin-polarization impact on fusion neutronics.

Lastly, focusing on OpenMC, the FREDA team developed capabilities for ease of use in coupled fusion device modeling and simulation. First, the OpenMC depletion (activation) solver capability was streamlined to a standalone activation solver, and this solver was validated against the Fusion Neutron Source (FNS) activation benchmark. From the results, an additional capability was developed to add multigroup-flux-weighted reaction yield fractions to OpenMC for a more accurate activation calculation. Lastly, a workflow was developed to generate tetrahedral mesh tallies for OpenMC and conversion of the result for use in other tools like OpenFOAM.

In conclusion, through the FREDA project, we aim to narrow the gap between reality and fusion neutronics modeling and simulation. With the commercial reactors having an elevated operating domain in terms of temperature, operating time, and neutron fluence, we expect the need for tighter coupling between neutronics, activation, and thermal hydraulics. By developing the capability to explore these uncertainties of modeling and simulation, we hope to provide the fusion neutronics community with a valuable tool.

ITER relevant water activation experiments at the JSI KATANA facility

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As the primary coolant, water is used in most of today's fission reactors and is also one of the most promising coolants for future fusion reactors. During the cooling of the reactor in-vessel components, the water is exposed to the high energy (E ~ 14 MeV) neutrons produced in the fusion reactions. It is activated and produces radioactive nuclides that release 6 MeV – 7 MeV gamma rays and delayed neutrons (in the energy range 0.4 MeV -1.7 MeV). The prediction of the dose rate field due to activated water is a complex task that requires the coupling of computational fluid dynamics (CFD), radiation transport and activation.

As part of the Preparation for ITER Operation (WP PrIO) work package, EUROfusion is addressing this issue in two ways: by developing appropriate computational tools using different approaches and by developing experiments and measurement methods to validate the codes.

There was a need for a coupled CFDand activation fluid simulation, so several tools (FLUNED, RSTM, GammaFlow, ActiFlow) were developed by different teams, each focussing on their specific strengths. In the case of F4E, RSTM was developed using Ansys Fluent, the ITER reference code for CFD calculations. As this is proprietary software more suited to industrial applications, the research organisations such as UNED/UKAEA used the most versatile open source alternative OpenFOAM as part of their existing workflows. It is important to have different solutions based on both proprietary and open-source codes. Such an approach enables a code-to-code comparison of performance and results, cross-validation and can take into account different ITER operating scenarios

In 2023, a closed water activation loop called KATANA was commissioned at the TRIGA Mark II research reactor of the Jožef Stefan Institute (JSI TRIGA) in Slovenia, which serves as a well-defined and stable source of high-energy gamma rays and neutrons from activated water. The KATANA facility can operate in pulse or steady state mode, has an adjustable flow rate and also features a delayed loop.

The KATANA water activation loop can be used to test radiation detectors, to perform shielding studies and, most importantly, for benchmark experiments to validate the newly developed codes. The KATANA loop was developed and optimised to achieve the highest possible water activation, which enables the measurement of neutrons from activated water. In addition, an ITER-relevant irradiation head was designed to reproduce the water flow and activation characteristics more similar to those expected in ITER.

In 2024 three experimental campaigns were performed, in which neutron and gamma-ray fields around the KATANA facility were measured using various active and passive neutron and gamma-ray detectors. As for gamma spectrometers HPGe (High Purity Germanium) and $LaBr_3$ (Lanthanum Bromide) were used, while gamma sensitive TLDs (thermo-luminescent dosimeters) were used to assess the gamma dose. As for neutron detection a complex detector based on an array of ^{3}He counters and two Lithium-glass scintillators were exposed to the radiation field, as well as activation foils of Au, Ni and In. Gamma rays from ^{16}N and delayed neutrons from ^{17}N were successfully detected and measured and such experimental data set, together with uncertainty analysis, will serve as benchmark experiments for validation and improvement of fluid activation codes.

Neutron-Based Characterization of the KATANA Activation Loop Using JSI TRIGA Reactor Pulsed Operation

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KATANA is a water activation facility located in the Jožef Stefan Institute TRIGA Mark II reactor in Ljubljana, Slovenia. Since its commissioning at the end of 2023, it has completed three experimental campaigns that have contributed to a better understanding of the water activation processes and their modeling, which are important in the context of research in nuclear fusion, in particular for the future operation of the ITER machine. The facility is designed to provide a flexible benchmark-quality environment for the validation of computational tools for the activation of water and corrosion products. It will also support shielding experiments, in particular for ITER-relevant materials, and serve as a stable source of high-energy gamma rays and neutrons.
The computational tools developed for ITER cooling loop characterization consist of two primary methods: computational fluid dynamics (CFD) in conjunction with neutronic simulations. These methods are crucial for the accurate modeling of distributed radioactive sources, such as the cooling water in future fusion plants. CFD focuses on detailed modeling of the thermal hydraulics, while neutronic simulations are used to predict the behavior of neutrons and their interactions. By integrating these two approaches, a more accurate and comprehensive understanding of water activation under different operating conditions can be achieved. Usually, these calculation tools are validated by dose rate measurements at different positions around the radiation source to ensure the accuracy and reliability of the simulations.
This approach is holistic, as the measurements combine neutron and water flow information, which cannot be separated from each other. In the present study, a novel method was developed to characterize the water activation loop using neutron detectors during operation of the JSI TRIGA reactor in pulse mode. Pulse operation allows the irradiation of a finite amount of water in the inner end of the KATANA water activation loop in a short length of time, typically below 1 s. As the irradiated water circulates through the loop, it emits neutrons along the way; by measurements using neutron detectors around the loop, it is possible to track the flow of the volume of activated water and infer information on the flow dynamics in the system. This approach provides valuable information on the KATANA device characteristics and enables increasing the accuracy of experimental data and improving predictive models for future fusion reactors.

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Characterization and Applications of Accelerator-Based D-T Neutron Sources: Insights from Recent Experiments

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The Institute for Plasma Research (IPR) at India has set up an accelerator based D-T/D-D neutron generator facility with a neutron yield of 10^{12} neutrons per second. The neutron generator operates using a deuterium ion (D+) accelerator, which targets a solid tritium target, producing neutrons through fusion reactions. This facility presents a safer alternative to traditional fission reactors, offering the advantages of high neutron flux and a broad energy spectrum all under a one roof. The versatility of this neutron source paves the way for a multitude of research applications, particularly in the fields of fusion neutronics such as nuclear cross section measurement, electronics & sensor testing, TBM mock up experiments, nuclear science, and space technology. Recent experiments at the neutron irradiation facility at the IPR focused on testing electronic components like optocouplers, field-effect transistors (FETs), static random-access memory (SRAM), analog-to-digital converters (ADCs), and instrumentation amplifiers (INAs) with 14-MeV neutrons to understand radiation-induced damage. Performance evaluations after each step of irradiation showed the optocoupler partially damaged at 5.31×10^{11} n/cm² and completely damaged at 1.77×10^{12} n/cm².This neutron generator facility offers an optimal setting for testing electronic components designed for high-radiation environments, such as those found in nuclear reactors and space satellites. In another set of experiments, materials such as P91 steel, Reduced Activation Ferritic Martensitic Steel (RA-FMS), and antimony (Sb) and zirconium (Zr) were irradiated with fusion neutrons. Activation analysis was conducted to study the properties of these materials under neutron exposure and measurement of the neutron-induced reaction cross-section for Sb and Zr. A significant ongoing development is the creation of a multipurpose moderator system that can utilize both fast and thermal neutrons. This system is designed to replicate the neutron energy spectra outside the main vessel, akin to those found in ITER-like fusion reactors, thereby facilitating the testing of critical components and sensors.

The work highlights IPR's capabilities in fusion neutron irradiation and its diverse applications. It presents the latest findings from fusion neutron experiments and details the design features of the multipurpose moderator system, which generates various energy spectra for experimental purposes. Additionally, it outlines the different setups available for component testing and other experiments conducted at the facility. The commissioning of this neutron generator facility signifies a major milestone for IPR, opening new avenues for research and innovation in India.

Methodologies and uncertainties in the determination of activated corrosion products in the ITER WCLL TBM water cooling and purification systems

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	Oral Contribution	☒

Accurate estimation of activation corrosion products (ACP) is of paramount importance in the nuclear field. Experience in pressurized light water reactors (PWR) demonstrates that ACP contribution to occupation radiation exposure (ORE) during plant operation and maintenance is up to 85% of worker exposure. In powerful magnetic confinement fusion experiments like ITER, harsh and peculiar (i.e., with significant differences with respect to fission plants) radiation environment poses new challenges in the estimation of ACP for the safety analysis of maintenance and accidental condition (e.g., loss of coolant accident - LOCA). More specifically, the materials employed in large tokamaks exposed to the strong neutron field of the D-T plasmas are subject to unprecedented radiation-induced damage, orders of magnitude larger than in fission reactors. Dedicated steels such as EUROFER97 have been developed to reduce activation under neutron fluence, at the expense of weak corrosion-resistant properties, resulting in an increased interest in ACP circulation and deposition.

The goal of this presentation is to provide an overview of the modeling methodologies applied in the determination of the ACP contribution, including a critical review of the typical approximations and simplifications employed and their potential impact. The uncertainties on the input parameters will be discussed including their influence on the final estimations. As a test case, the results of the activities performed in the context of the ITER WCLL TBM water cooling and purification system with the OSCAR-Fusion code will be presented. This system is composed by large number of parts that result in a big model of 113 equivalent regions, that are linked to components placed from the zones just facing the plasmas (e.g. the TBM set) to the outer port cell.

From the neutronics point of view, several points should be carefully considered. The OSCAR-Fusion code implements a zero-dimensional control-volume approach with a time solver most suited for corrosion calculations. Since the mechanics of the activation in a pulsed fusion machine may account for extreme power oscillations, relevant inputs and on-purpose selection of options are requested. The use of supporting MCNP and FISPACT calculations will be shown as well as the methodologies employed in the determination of operational scenarios, compatible with fast response times in OSCAR-Fusion but still sufficiently accurate. The effects of the reduced neutron fluence proposed ITER baseline (DT1) as well as the model definition for the representation of the overall campaign as a series of long pulses will be presented as well as its relationship with the other material and corrosion parameters.

Investigation of high-energy background in gamma-ray measurements for fusion power determination in DT plasmas

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The fusion power measurement in a tokamak is essential for the control of operations as well as licensing and research. Traditionally in DT fusion plasmas, fusion power is measured by counting neutrons produced by the $T(D,n)^4He$ reaction. However, a novel method based on the spectroscopy of gamma-ray produced by the $T(D,\gamma)^5He$ reaction has been successfully demonstrated at JET[1,2]. While neutron diagnostic requires the calculation of the adjoint flux using an accurate MCNP model of the reactor, gamma-ray diagnostic only requires knowledge of the plasma volume in the detector line of sight and of the plasma spatial distribution: furthermore, it does not require extensive in-vessel calibration campaigns.

This technique is very promising but it presents several challenges: among these, there is the need to develop an appropriate neutron attenuator, which is mandatory since the $T(D,\gamma)^5He$ reaction branch is $2.4\cdot10^{-5}$ less likely than the $T(D,n)^4He$ branch[2]. Another challenge is to obtain detailed simulations of background, especially in the energy range between 15 and 20 MeV where the 16.7 MeV peak of the $T(D,\gamma)^5He$ reaction is present. To properly understand what is causing the background in this high-energy range beam-on-target experiments were performed at the Frascati Neutron Generator (FNG). By performing MCNP simulations of the experiment we determined the origin of the high-energy background to be caused almost entirely by the steel and copper in the proximity of the target. The findings are consistent with previous works found in literature [3]. Furthermore, we extended this study to the environment of the JET tokamak, by developing an MCNP model capable of describing the high-energy background of the JET tangential gamma-ray spectrometer.

Due to the low probability of these events, several variance reduction techniques, such as Weight Windows, DXtran and F5 tallies are applied. Moreover, different nuclear libraries have been used and the results of this comparison together their validation against experimental results will be reported. Finally, some studies have been performed to compare different materials and alloys that, by reducing the high-energy background, could be of help in the designing of a gamma-ray diagnostics suitable for fusion power measurements in future DT machines.

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Neutron Flux in IFMIF-DONES operations rooms: a comparison between Monte-Carlo and Deterministic code predictions.

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The main requirement for IFMIF-DONES [1,2] is the production of a neutron flux of sufficient intensity to allow accelerated testing of materials beyond the expected operational lifetime of DEMO, with an irradiation volume large enough to allow the characterization of the macroscopic properties of materials of interest for the engineering design of the reactor.
The facility will generate neutrons with an energy spectrum similar to that of a DT (deuterium-tritium) fuelled nuclear fusion reactor. A commercial fusion reactor will require materials capable of withstanding 150 DPA, which is currently an unexplored region. In IFMIF-DONES, neutrons will be generated by bombarding liquid lithium with accelerated deuterium ions (deuterons), in quantities sufficient to cause damage equivalent to that in the first wall of a fusion power plant.
To ensure that the design of this facility meets the required safety standards, numerous experimental tests and analytical and numerical calculations through simulations using appropriate tools are being carried out. Regarding the irradiation doses in the collimator and adjacent rooms, the main numerical analysis is carried out using MCNP [3], while the Fusion Group at the Barcelona Supercomputing Centre has been invited to provide additional calculations as an alternative and validation to those performed with MCNP by the Technical Commission experts. For this work, we use a computational tool called NEUTRO, which is a deterministic neutron transport code based on the Boltzmann transport equation and integrated into the Alya framework [4,5]. The solver uses the Discrete Ordinates Method in the angular direction, multi-group for the energy discretisation and the Finite Element Method (FEM). It has been tested against various experimental, analytical and numerical benchmarks [6]. We present the NEUTRO results in comparison with the MCNP analysis of the neutron flux in the collimator rooms and the operations room.

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Extended Adjoint Driven Importance Solution (XADIS)

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Please, indicate type of contribution:	Poster session	☐
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SPARC is a compact, high-magnetic field tokamak being designed and constructed by Commonwealth Fusion Systems. The neutron diagnostics system in SPARC is the main tool to calculate fusion power and assess plasma burning conditions. The design of these neutron diagnostics relies on radiation transport simulations to optimize detection efficiency through the selection of materials and design of geometries. Of particular importance is the prediction of the interference in detector responses due to scattered neutrons from the different tokamak components as well as structural materials between the tokamak and the detectors. In principle, the importance of scattered neutrons to a detector can be calculated using standard Monte Carlo simulations; however, due to the slow convergence and limitations of available Monte Carlo codes, quick design iterations and parametric studies become difficult. To address these shortcomings, we have developed a framework that uses deterministic transport and contiguous unstructured mesh models to determine the scattering contribution of the different components in the model to a detector response. The tool is based on contributon theory and evaluates the related mathematical operations on spherical harmonic moments using the macroscopic multigroup cross sections, and the adjoint and forward fluxes generated by the Attila deterministic solver. Tests were performed to verify the initial implementation of the code. These tests were carried out using simple models and the results were compared against MCNP data processed through the PTRAC capability. The comparison shows reasonable agreement between the MCNP results and those produced by the new tool. Future work includes conversion to a higher performance production implementation for improved run times and creation of a visualization tool.

IFMIF-DONES related activities at Lithuanian Energy Institute

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Please, indicate type of contribution:	Poster session	☒
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Lithuanian Energy Institute (LEI) actively contributes to the IFMIF-DONES (International Fusion Materials Irradiation Facility – Demo Oriented NEutron Source) project as a member of the EUROfusion consortium, primarily through the dedicated Early Neutron Source (ENS) work package. IFMIF-DONES is a critical accelerator-based facility designed for the testing and qualification of materials for future nuclear fusion power plants and installations.

As part of its involvement, LEI has conducted extensive particle transport and radionuclide inventory analyses to support the safety and design optimization of the IFMIF-DONES facility. Key contributions include detailed neutronic analyses of essential testing components such as the High-Flux Test Module (HFTM) and the Start-Up Monitoring Module (STUMM), as well as comprehensive neutronic and radiological assessments of structural materials. Additionally, LEI has studied the most activated areas within the IFMIF-DONES building, including the Test Cell, Lithium Interface Cell, and Lithium Loop Cell, among others.

LEI has also investigated particle streaming and shielding optimization. These activities consist of evaluations of radiation barriers, as well as assessments of localized shielding challenges, such as neutron streaming through bolt mounting cavities and other small structural features. Through these contributions, LEI plays a significant role in enhancing the safety and performance of the IFMIF-DONES facility supporting the development of sustainable nuclear fusion energy.

Delayed neutron measurements from activated water at the JSI KATANA facility with ^{3}He and Lithium glass neutron detectors

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Neutron-induced water activation is a major concern in nuclear facilities. In nuclear fusion environment, this issue is well known and has been considered in ITER since the early stages of the project. It represents an additional radiation source, increasing both personnel exposure and the load on machine components.
Neutrons with energies exceeding 8 MeV can activate oxygen in water, leading to the production of high-energy gamma rays from ^{16}N via the $^{16}O(n,p)^{16}N$ reaction and delayed neutrons from ^{17}N via the $^{17}O(n,p)^{17}N$ reaction. Energies of neutron produced from the latter reaction are 0.386 MeV (37.7%), 1.16 MeV (49.8%), and 1.69 MeV (6.9%) with a half-life of $T_{1/2}$=4.173±0.04s. Although the natural abundance of ^{17}O is only 0.04%, the resulting delayed neutrons could become a significant radiation source due to the large volume of water expected in cooling loops. Therefore, characterizing the neutron field generated by water activation is of fundamental importance for ITER and future water cooled fusion reactors.
In the frame of the EUROfusion Work Package Preparation of ITER Operation (WP PrIO), three experimental campaigns were conducted in 2024 at the KATANA closed water activation loop facility, located within the TRIGA Mark II research reactor at the Jožef Stefan Institute (JSI) in Ljubljana. These campaigns involved several European research groups, who investigated various aspects of water activation.
This work presents ENEA's contribution to the latest KATANA experimental campaign, conducted at the end of 2024, focusing on the measurement of delayed neutrons from activated water. To characterize the neutron field, two different types of detectors were employed: the JCC-15, manufactured by Canberra, an array of ^{3}He detectors embedded in a large polyethylene cylinder, and the GS20 detector, produced by Scintacor, a compact lithium-6 enriched glass scintillator. Various measurements were performed using, in parallel, both detectors at different power of the TRIGA reactor, modifying the water flow rate within the KATANA loop, and analyzing different pump transient conditions.

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Supply and demand of tungsten in a fleet of fusion power plants

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Please, indicate type of contribution:	Poster session	☐
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To scale nuclear fusion power plants to commercial and global levels of adoption, the industry must be supported by a robust supply chain. For the use of tungsten as the primary element in plasma facing and radiation shielding applications in both spherical and D-shaped tokamaks, this requires that mining, processing, and manufacturing industries must be capable of supplying the tungsten quantity and quality required to enable systematic reactor construction and maintenance. ARIES-ST and EU-DEMO1 design points were used as the basis for a spherical and D-shaped tokamak, respectively, for the neutronic model calculations, to determine the replacement of tungsten components (plasma facing and radiation shielding) during a nominal forty (40) full-power-year (fpy) operation. ITER Grade W was selected for the plasma facing material and four radiation shielding material options, ITER Grade W, monolithic tungsten carbide (WC), tungsten boride (W2B), and tungsten carbide with binder (WC/Co), to investigate different material options. The central column radiation shielding material was found to be the main contributor to the consumption of tungsten in spherical tokamaks due to the proximity to the plasma source (i.e. neutron source) which reduces the fpy lifetime of the component. The average tungsten consumption for all designs over 40 fpy is ~4088 tonnes per reactor. Three distinct scenarios on tungsten supply were considered, presented and discussed, where tungsten sources were restricted by procuring from certain countries to emulate future potential scenarios on the tungsten market. The results of all three modelled scenarios reveal that new mining resources must be brought online by the mid-2040s to enable a sustainable and cost-effective tungsten supply for a fleet of fusion power plants by 2060. The results show that by 2060 U.S. Department of Defence 'restricted tungsten' production would have to reach ~31,000 tonnes per year, 2.38 times increase from today, simply to accommodate for the demand for the fusion industry. If the United Kingdom or United States construct and operate a fleet of fusion power plants up to the year 2100 and with no domestic supply of raw tungsten, then both countries could not use tungsten for the plasma facing and radiation shielding components as the supply does not meet demand without any intervention.

Development and management of reference IFMIF-DONES neutronics document: Nuclear Analysis Handbook

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	Oral Contribution	☒

IFMIF-DONES (International Fusion Materials Irradiation Facility – Demo Oriented NEutron Source) is a neutron source designed to irradiate candidate materials to be used in future fusion power reactors such as DEMO (DEMOnstration power plant). As part of the facility design, the impact of neutron field should be evaluated to demonstrate nuclear safety. In order to have a reference document with the most essential information from nuclear analyses, the Nuclear Analysis Handbook (NAH) has been developed. Having a common and updated entry point to the most relevant nuclear analysis is a key step.

The NAH collects system and subsystem analyses for the provision of nuclear responses for design and safety needs. It summarizes and organizes over 100 technical documents and reports in a system-oriented structure, following the facility Plant Breakdown Structure (PBS), and containing concise information on beam-on and beam-off doses, activation inventory, nuclear heating, neutron and gamma flux spectra, material damage, and so on. Also, it includes information about the most up-to-date neutronic models and commissioning and it helps to identify the data gaps in the facility design.

A critical aspect of NAH was the development of an automatized strategy to compare periodically the reports considered in NAH and the new ones. For that purpose, it has been developed a methodology to compare the documents stored in IFMIF-DONES repository, the EUROfusion IDM platform, with the reports considered in NAH, identifying both new and updated (new version) documents. This strategy is essential for a periodic update of NAH.

Study of the ITER Activated Corrosion Products: from neutronics to corrosion and contamination analyses and experiments

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The candidate materials proposed for the fusion cooling systems might release metallic cations and corrosion products (CP) whilst exposed to high-temperature and high-speed water. Moreover, under the expected neutron irradiation field, the CP can become activated, thus forming so-called activated corrosion products (ACPs). While assessing ACPs in fission reactors relies on well-established codes (*e.g.*, OSCAR from CEA), the formation of ACPs under ITER-like conditions still needs proper validation. The current codes should be able to account for the different neutron irradiation fields, the presence of strong magnetic fields, and the usage of further structural alloys (*e.g.* CuCrZr). Therefore, this work aim to support the validation of computational codes for calculating fusion-specific ACPs, mainly through benchmarking experiments producing fusion-relevant data. In this talk, an overview of the current activities supporting the EUROfusion work package PrIO (Preparation for ITER Operations) is discussed, including the construction of a dedicated ACPs water testing loop at the ENEA 14-MeV Frascati Neutron Generator, preliminary results obtained from corrosion experiments on CuCrZr samples performed at RINA, and collection of data from literature on CuCrZr, SS316L-IG and Eurofer97 steels. The literature and experimental data are being used to obtain a preliminary innovative corrosion law for the fusion materials for the ACPs codes development; this will support the assessment and consequent optimization of the Occupational Radiation Exposure (ORE), waste management and maintenance plans for ITER.

SIMULATION OF RADIATION BACKGROUNDS FOR THE SOUTH POLE NEUTRON TIME-OF-FLIGHT DETECTOR AT THE NATIONAL IGNITION FACILITY

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

The National Ignition Facility (NIF) at Lawrence Livermore National Laboratory is the world's largest and most powerful laser system for inertial confinement fusion. The NIF is a 192-laser beam facility capable of producing 2.2 MJ, 500 TW of ultraviolet light. Neutron time-of-flight (nToF) detectors are fielded in the NIF to measure neutron yield, ion temperature, and downscattering in the cold fuel for D-T implosions. Anisotropically assembled cold fuel may generate different nToF data when measured by detectors located at the NIF Target Chamber (TC) equator and poles. These measurements have played an important role in achieving MJ yields and target gain >1 at the NIF. The nToF suite at the NIF consists of five spectrometers positioned around the TC at distances of 18 to 24 m from the Target Chamber Center (TCC). A collimated nToF line of sight has been fielded near the TC South Pole (SP) to examine any possible anisotropy in the cold fuel. The SP nToF detector is located in the lowest floor level of the NIF's Target Bay (TB) and at a distance of ~ 18 m from the TCC. The detector utilizes a solid bibenzyl scintillator coupled to four detectors (a photodiode and three photomultiplier tubes (PMTs) with varying gain), as well as a Cherenkov based system. The line of sight includes a port collimator that is attached to the TC and a bore hole collimator in the TB concrete floor above the detector. In addition, a beam line get lost is constructed in the lowest TB floor to minimize backscatter at the detector location. Initial measurements indicated the need for installation of additional shielding to eliminate gamma backgrounds during the period before arrival of 14.1 MeV neutrons to the detector. Gamma arrival during this time period interferes with measurement of the "reaction-in-flight" (RIF) neutrons. The RIF neutrons can have energies up to ~ 30 MeV and are produced by up-scattered deuteron or triton undergoing D-T reaction with thermal ions A detailed MCNP model has been developed to simulate the expected neutron and gamma backgrounds at different locations in the NIF facility. The model includes most of the major components inside the TC and the major diagnostics and structures inside the TB. A set of MCNP Monte Carlo simulations with the full TB model were conducted to provide an estimate of the expected neutron and gamma backgrounds during D-T shots. A new shielding scheme was designed to reduce the gamma background. Measurements taken following the shield installation showed that the gamma background during the period of concern was reduced by about an order of magnitude.

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Activation analysis on the DEMO divertor materials: study of the impact of chemical composition and operational scenario

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The DEMO divertor is devoted to carry away the exhaust power and remove impurities generated in the plasma. Therefore, it is exposed to significant thermal power and nuclear loads, i.e. nuclear heating, gas production and transmutations, radiation damage and induced activation of the irradiated materials. In particular, activation issue represents a demanding concern to be addressed for the operative phases, waste management and safety aspects.

This study is focused on an extensive inventory and activation analysis which is carried out within the EUROfusion DEMO divertor Work-Package. The latest design of DEMO divertor has been considered and, an equivalent three-dimensional MCNP heterogeneous model has been generated and integrated in the reference DEMO MCNP model with the Water Cooled Lithium Lead (WCLL) blanket configuration. The neutron spectra calculated with MCNP5 Monte Carlo code have been used for a fully inventory calculation by means of FISPACT-II aimed at assessing the impact of chemical composition of relevant divertor materials (e.g., Eurofer, AISI SS316Ti, CuCrZr, Inconel-718, Tungsten, etc) and irradiation scenario on specific activity, contact dose rate and decay heat after plasma shutdown. Indeed, a dedicated sensitivity study has been performed varying within a reasonable range of values the concentrations of impurities such as Co, Ta and Nb etc and the operational scenario. These results can provide significant outcomes in the materials selection as well as in the planning of the safety and waste management of the DEMO reactor.

The main results of this study are presented and critically discussed.

Machine learning for the design of GENeuSIS: a neutron test bed facility for diagnostics and critical components of ITER

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

The radiation environment in ITER and high-performance fusion devices exhibits significant variability in space, time, energy, and angle, impacting materials and components differently based on the energy spectra distribution. However, current neutron facilities cannot replicate the full complexity of these energy spectra, making comprehensive experimental testing under ITER-like conditions unattainable.

GENeuSIS (General Experimental Neutron System Irradiation Station) introduces an innovative, modular, and flexible test-bed concept designed to replicate specific neutron and gamma energy spectra distributions. GENeuSIS aims to create tailored irradiation conditions in an assembly made by different common materials. Recent MonteCarlo simulations have focused on replicating neutron energy spectra relevant to ITER locations using the 14 MeV neutrons generated by the Frascati Neutron Generator (FNG).

Conceptual designs, validated through MCNP Monte Carlo simulations, demonstrated the feasibility of GENeuSIS for ITER applications. Two assemblies were designed: GENeuSIS-I, replicating the neutron spectra in the Port Interspace, and GENeuSIS-II, tailored to the Port Cell of an ITER Diagnostic Equatorial Port.

To speed-up the design process, a machine learning model based on neural networks is under development, using data generated from the simulations. This model aims at accelerating the pre-analysis by identifying optimal material combinations and configurations to reproduce desired neutron and photon energy spectra.

This work represents the first step in integrating machine learning into GENeuSIS, highlighting its ability to efficiently predict neutron fluxes based on a given material configuration, but it also gives an overview of the next phase that focuses on the inverse process: predicting optimal material configurations to achieve specific desired flux profiles.

Accelerating and Optimizing Neutronic Simulations with Differentiable Neural Operator-based Surrogate Models

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	Oral Contribution	☒

The IFMIF-DONES project, which aims to pave the way to understanding the effects of high energy and flux neutronic irradiation on materials, is a crucial step for the future of fusion reactors. Traditionally, Monte Carlo simulations have been employed to study neutronic transport and interactions; however, their computational cost hinder the ability to optimize and control elements of the environment, such as geometry or accelerator configuration, in the case of IFMIF-DONES. In this work, we propose an approach based on Deep Learning Surrogate Models, which employes a Neural Operator known as MIONet to predict neutron-related tallies such as flux, damage energy, H and He production, and heating.

By using a custom differentiable version of the MIONet, we enabled the ability of, following the predictions, optimize the initial Gaussian parameters that define the deuteron footprint on the lithium curtain. Several optimization techniques are tested, including Gradient Descent, Bayesian Optimization and Multi-Objective Evolutionary Optimization, allowing for a more efficient parameter space exploration and the creation of multidimensional Pareto Fronts based on customizable optimization functions.

The results of this work demonstrate that these models achieve speedup factors of around 10^6, compared with Monte Carlo simulations, with Coefficients of Determination (R^2) above 0.986 for every model, up to 0.996. Additionally, the integration of optimization techniques allows for the refinement of beam characteristics and irradiation patterns, with applications in inverse design where desired outcomes are set, and the model is utilized to discover optimal input parameters.

Through this work, we highlight the computational efficiency achieved by these models, as well as their potential for advancing real-time control systems and design optimisation within fusion facilities. Future work will focus on refining the 3D High Flux Test Module model according to the improved configurations, integrating more parts of the system, such as the accelerator, to explore integral control of the device, and updating the model's architecture to achieve its most efficient configuration.

Neutronics Studies in Korea for KODA Diagnostics, IBTF, and D1S Code Development

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	Oral Contribution	☒

A comprehensive range of neutronics studies has been conducted for various projects, including the KODA diagnostic systems, the Integrated Breeding Test Facility (IBTF), and the development of the D1S code. For the KODA diagnostic systems, detailed evaluations have been carried out to analyze key neutronics parameters such as nuclear heat generation, and ShutDown Dose Rate (SDDR). These assessments were performed for multiple components, including upper port #18, the vacuum ultraviolet spectrometer system, and the neutron activation system, using SRC-UNED, FENDL-3.1, TopMC, SpaceClaim, and C-model (R181031). Regarding the IBTF, nuclear analyses have been conducted for the development of a conceptual shielding design, considering both prompt radiation and SDDR. Additionally, the optimization of the tritium breeding unit has been performed. These evaluations were carried out using MCNP6 and FISPACT-II. Additionally, the D1S code is being developed as part of an ongoing effort to enable comprehensive SDDR assessments for both fusion blanket systems and Generation IV fission reactors. The current progress and findings from these neutronics studies will be presented.

Remarks on beryllium nuclear data for fusion reactors

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

Beryllium, ^{9}Be, is an important functional material for a fusion blanket as a neutron multiplier and moderator and will be irradiated in test modules at the Advanced Fusion Neutron Source (A-FNS) to obtain the neutron irradiation characteristics data. However, the nuclear data of beryllium has been suspicious by several integral experiments.

JAEA/FNS carried out an integral experiment with a thick beryllium assembly with DT neutrons 30 years ago. It has been reported to date that analyses of this experiment largely overestimate the measured data sensitive to low energy neutrons even if the latest nuclear libraries are used, though they rather agree with the measured data sensitive to neutrons above 1 MeV.

Recently, the European Spallation Source group produced new thermal neutron scattering law data of beryllium with different crystallite sizes, whose cross sections were different due to the extinction effect. In this study, we adopted the thermal neutron scattering law data of beryllium with different crystallite sizes for the analysis of the JAEA/FNS beryllium experiment with MCNP-6.20 and JENDL-5. As a result, the overestimation of the measured data sensitive to low energy neutrons decreased with increasing crystallite sizes. It demonstrates that the thermal neutron scattering law data of beryllium without the extinction effect cause the overestimation.

In this study, we also found another issue on beryllium. The calculated reaction rates of the $^{115}In(n,n')^{115m}In$ reaction sensitive to neurons above 350 keV are clearly different among JENDL-5, JEFF-3.3 and FENDL-3.2c. We also examined this issue and specified that the secondary neutron spectrum in the (n,2n) reaction caused the discrepancy.

The secondary neutron production data in the (n,2n) reaction of ^{9}Be should be revised and the thermal neutron scattering law data of beryllium considering the extinction effect should be added to the next FENDL.

An overview of UKAEA's current fusion neutronics and radiometric activities

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	Oral Contribution	☒

UKAEA has a rich history in fusion neutronics, radiometics and, neutron and gamma diagnostics; establishing themselves at the forefront of the scientific fields. This talk presents an overview of UKAEA's current activities, including methods and workflow developments, recent computational analysis for various internal and external programmes, developments and applications of the UKAEA's Radiological Assay and Detection Lab (RADLab), and developments in the area of neutron and gamma diagnostics.

The talk will also touch on a number of current UKAEA programmes with relevance to the fusion neutronics community including LIBRTI, JET, MAST-U and STEP. The Lithium Breeding Tritium Innovation (LIBRTI) is a new programme to advance fusion fuel advancements, where a 14 MeV neutron source will be installed at UKAEA's Culham campus with the initial aim of demonstrating controlled tritium breeding. MAST-U is the UK's current spherical tokamak while STEP (Spherical Tokamak for Energy Production) is the UK's programme to develop a proto-type fusion energy plant. The Joint-European Torus (JET) is currently being decommissioned after 40 years of service yet still offers a wealth of data in its decommissioning processes.

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Uncertainty quantification and recent inboard studies in STEP

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

The Spherical Tokamak for Energy Production (STEP) is a significant UK endeavour to build a prototype fusion power plant that will demonstrate net energy and fuel self-sufficiency, with a target of 2040. STEP will utilize a range of novel technologies, including High Temperature Superconducting magnets and breeder blankets, which will be subjected to harsh radiation environments.

During the early design phases, a simple neutronics model is required that can perform rapid, accurate and reliable calculations to inform on design decisions. However, there are multiple sources of uncertainty within the model, particularly at such an early stage. Ideally, experimental campaigns would be conducted to investigate the response of these novel technologies to expected neutron and photon fluxes to reduce the uncertainties and risks associated with the overall performance and lifetime. However, it is difficult to represent the STEP radiation environment and operating conditions anywhere outside of a fusion power-producing reactor, which does not currently exist.

For the project to be able to make informed decisions at these early design phases, it is critical to combine uncertainties to understand the overall confidence we have in the predicted performance and lifetime metrics. A description of STEP's inboard midplane model and its workflows are presented, along with recent studies, a description of the quantification of uncertainty and how this combination has informed on significant design decisions of STEP.

F4Enix, a new Python API for pre and post processing of MCNP inputs and outputs

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Please, indicate type of contribution:	Poster session	☐
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The reliable computation of nuclear responses for fusion reactor and components are a complex and resource-intensive process. It encompasses the preparation of exceptionally large and detailed computer models that could not possibly handled by hand and need a high degree of automatization to be dealt with.

To address these kinds of challenges, a great number of scripts was produced during the years at Fusion for Energy (F4E) that presented a wide range of maturity, verification level and documentation quality. The neutronics team of F4E recently decided to collect all these codes, refactor them, generalize them, and aggregate them into a unique pip installable python package named F4Enix distributed under the EUROPEAN UNION PUBLIC LICENCE v. 1.2 terms. The primary objectives of F4Enix are:

- Automate and streamline the pre and post-processing operations involved in nuclear response computations for ITER or similar projects.
- Significantly enhance the efficiency, capability, and overall quality of the entire nuclear analysis workflow.
- Allow to maintain, test, and document all code related to nuclear analyses workflows in a single place.
- Build synergies between the different code parts.
- Provide a useful tool to F4E suppliers.

To foster collaboration, encourage improvement, and avoid duplication of efforts, the development of F4Enix follows an open-source approach. To ensure the quality and reliability of F4Enix, the development team follows best practices such as continuous integration, code review, testing, and extensive documentation. F4Enix uses GitHub as a platform for hosting the source code, managing issues, and collaborating with other developers. F4Enix also provides a comprehensive and up-to-date documentation that describes the installation, usage, and API of the package.

An open-source hybrid unstructured mesh—CAD fusion multiphysics analysis workflow in FENIX

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

Plasma facing components in fusion devices will endure extreme neutron and heat fluxes. To facilitate their design using simulation tools, the open-source Fusion ENergy Integrated multiphys-X (FENIX) framework is being developed to model these components with a high-fidelity multi-physics multi-dimensional approach. It can iteratively resolve couplings between all the physics at play, from neutron radiation, to thermomechanics and near-wall plasma dynamics. This framework is based on MOOSE, the Multiphysics Object Oriented Simulation Environment, which was developed and continues to be advanced through ongoing collaboration among the National Laboratories of the United States since 2008, for advanced nuclear, geomechanics simulations and other applications. FENIX couples numerous simulation tools, including OpenMC through Cardinal, the Tritium Migration Analysis Program version 8 (TMAP8), the NekRS CFD software, and most MOOSE modules.

For the coupling of radiation transport and other physics, FENIX supports a hybrid workflow between Computer Assisted Design (CAD) and unstructured mesh geometries—crucial for representing complex geometries. The CAD can be generated from skinning the unstructured mesh, enabling a coarse geometry for efficient particle transport, but still resolving the local material compositions and temperature gradients. Neutron transport is performed using DAGMC on the CAD through Cardinal, then tallied quantities (such as the heat deposition or the tritium generation rates) are mapped from the tally volumetric mesh tally to the other physics' unstructured mesh. This coupling was exercised on a simplified tokamak geometry, coupling neutron transport with the heat conduction equation, and on a monoblock divertor problem, coupling additionally with tritium migration. Mesh convergence studies highlight the importance of the mapping conservativeness. Coupling with thermomechanics is further enabled by the generalization of the approach to moving meshes. The presentation will include these coupled analyses as well as an update on status of the FENIX framework.

Development of the OpenMC workflow for ITER DNFM neutronics assessment

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Please, indicate type of contribution:	Poster session	☐
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MCNP [1] is the only code in use for neutron transport analysis at ITER. At the same time OpenMC [2] software package is a promising and even necessary alternative to MCNP. At least OpenMC code can be used for preliminary evaluations, lost particles elimination, weights calculations for variance reduction and other steps in the process of neutronics assessment. The code constitutes the powerful and convenient instrument for these tasks. The essential advantage of the OpenMC is its open-source status. This allows to adapt the code for specific tasks of each research group. The project is supported by more than 100 developers from all over the world. The pace of the code development is excellent, due to experienced team, sound engineering process and applied modern technologies. This work discusses the use of the OpenMC code for the ITER neutronics modelling, model translation between MCNP and OpenMC and several verification milestones.

The main problem of using OpenMC as an alternative to MCNP is the need to support equivalent models for MCNP and OpenMC. As for geometry, there are several solutions for forward and backward conversion between MCNP and OpenMC, as well as between MCNP, CAD and OpenMC. These solutions still need to be finalized and more rigorously tested, but essentially the models are well synchronized between the two codes.

For the fusion neutronics tasks, the problem of determining the DD or DT source for neutron transport calculation in OpenMC is solved by creating special software that allows for specifying of desired plasma source in three ways: in tabulated form; directly from the analytical model of plasma shape; translation from MCNP SDEF.

Having developed a geometric model and a plasma source using ITER DT scenario (500 MW), we performed calculations of neutron transport, heating, and fission rate for the DNFM diagnostics using OpenMC code. In addition, the results of the OpenMC calculation were compared with the results obtained earlier in MCNP as a way to verify OpenMC results. The obtained relative difference in the calculated values of the total neutron flux of no more than 12% and the difference in the calculated values of the uranium fission rate of no more than 15%. The biggest discrepancy in the results was observed in the thermal region of the neutron spectrum. In conclusion we propose further steps necessary for OpenMC benchmarking against other codes, as well as in live experiment with neutron source.

The work is supported with the state contract of 20.03.2024 № H.4a.241.19.24.1024 «Special equipment development, experimental manufacturing, testing and delivery preparation to meet Russian Federation's liabilities on ITER project in 2024 year».

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Overview of the Neutronics Calculations for EU DEMO at Lithuanian Energy Insitute

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Please, indicate type of contribution:	Poster session	☒
	Oral Contribution	☐

Interest in nuclear fusion is growing due to the increasing demand for sustainable, low-carbon energy sources. In the European Union, fusion research is coordinated under the EUROfusion program, which aims to enable fusion-based energy production. A key project in this effort is the European Demonstration Fusion Power Plant (EU DEMO), which bridges experimental reactors and commercial fusion energy. Neutronics calculations ensure EU DEMO's safety, efficiency, and feasibility, particularly in neutron transport and material activation. The Lithuanian Energy Institute (LEI) actively contributes to EUROfusion research, including conducting neutronics and inventory calculations for EU DEMO. This study presents key findings from these calculations and their implications for reactor design and operation.

The main findings are related to 5 constructional and 3 functional materials in EU DEMO components: vacuum vessel, divertor, and breeding blanket. For this calculation, two breeding blanket modules: Helium-Cooled Pebble Bed (HCPB) and Water-Cooled Lithium Lead (WCLL) were used.

The neutronics calculations focused on neutron transport, material activation, and radiological characterization of EU DEMO components. The study used advanced numerical models and nuclear cross-section libraries to evaluate neutron flux distribution, isotope generation, specific activity and decay heat. For the calculation of neutron spectra, the MCNP6 code was used with FENDL-3.2 nuclear data, while activation and decay heat calculations were conducted with FISPACT-II using the TENDL-2017 library. Activation calculations also determined the radioactivity levels and waste classification over time. A comparative analysis of HCPB and WCLL breeding blanket concepts was conducted to assess their impact on neutron interactions. These calculations support material selection, reactor safety, and long-term waste management planning.

Variance Reduction Capabilities Within a Cloud-Based OpenMC Interface

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Please, indicate type of contribution:	Poster session	☐
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Precise understanding of neutron radiation transport within complex systems is a critical aspect of fusion reactor design for tritium breeding, health and human safety, and shutdown maintenance concerns. CAD models are now frequently used for both engineering design and neutronics simulations. This higher geometric complexity comes at the cost of increased computational resources required to simulate neutron fluxes with low statistical uncertainty, especially in deeply shielded regions. Global Variance Reduction methods allow for biases to be introduced into models so that particle fluxes are more uniformly distributed and statistical uncertainties everywhere are minimized.

We will discuss recent modifications to OpenMC enabling two different approaches to generating weight windows maps for global variance reduction for complex models. One is an iterative approach to generating weight windows, known as the MAGIC method[1], where upper and lower bounds of weight windows are estimated through pilot simulations and ultimately converge after many iterations. Another is a novel implementation similar to the FW-CADIS framework utilizing deterministic methods. This solves weight windows values based on on the results of the Random Ray method, as originally demonstrated in the SCONE code[2].

Finally, we will show a browser-based interface for configuring these simulations, executing them using cloud computing resources, and interactively producing visualizations of the simulation results. This interface is convenient for production use, but can also be used by beginners and design engineers for exploration outside of a company or organization's primary neutronics workflow.

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Neutron shielding of low aspect ratio torii modeled by Monte Carlo methods and Machine Learning

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Please, indicate type of contribution:	Poster session	☐
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Liquid plasma-facing walls allow for increased neutron-wall loading, expanding the design space of fusion power-plants towards higher field, more compact tokamaks and stellarators subject to high neutron fluxes. This study specifies and models the materials for a compact blanket featuring a plasma-facing liquid wall. A selection of solid and liquid materials, including lead (Pb) and lithium-lithium hydride (Li-LiH), was previously shown to shield from neutrons, breed tritium, and extract heat from a 2.2 GWth deuterium-tritium fusion reactor, in a relatively compact total thickness of about 1 m. Monte Carlo simulations were carried out for this purpose in cylindrical geometry, i.e., in the limit of infinite aspect ratio *A* [V. Prost *et al*., *J. Nucl. Mater*. **599**, 155239 (2024)]. Calculations performed with the OpenMC Monte carlo code are presented here for torii of circular and non-circular cross-section. As expected, their inboard side is subject to a higher neutron flux and thus requires more shielding than the outboard side. Shielding layers should thus be excentric to the plasma cross-section. More importantly, this constrains the lowest *A* and smallest major radius *R* at which shielding is possible. Maps of possible *A* and *R* are generated, indicating that the central column of a spherical tokamak of A = 2 and R= 4 m can be shielded. Under the approximations made, stellarators of A = 4 and R = 4 m are even easier to shield with an overall inboard thickness of about 1 m is required, but higher geometrical fidelity is needed to confirm this. In preparation for those costly calculations, for optimization purposes and integration within a system code, we trained machine learning models for each toroidal geometry on several thousands of simulations. Typically, an OpenMC simulation took tens of minutes, while the surrogate model required only a hundred milliseconds. As a result, the tritium breeding ratio, energy multiplication, can now be rapidly predicted within a few percents and for DPA, neutron fluence within tens of percent. Finally, thickness needs are shown to only mildly increase with fusion power.

Neutronics Design of a Breeder Blanket for a Spherical Tokamak Fusion Pilot Plant

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

The low aspect ratio of spherical tokamaks increases the solid angle of the outboard breeding blanket and decreases the solid angle of the inboard central column. This geometric effect enables a viable outboard-only breeding blanket solution particularly when optimized with the choice of liquid lithium as the breeder material. This presentation will show how neutronics codes (MCNP and GEANT4) have been used to test the design space of breeder blankets and select the design.

This presentation will show tritium breeding ratio (TBR) and energy multiplication values for standard breeder designs and material inside spherical tokamaks and how things such as aspect ratio, elongation, penetration, limiters and ports affect the TBR.

To minimise the risks of lithium-6 enrichment and beryllium multipliers, Tokamak Energy has selected a liquid lithium breeder blanket. The design of the liquid lithium breeder blanket has been carried out using a combination of neutronics, computational fluid dynamics (CFD), magnetohydrodynamics (MHD), and mechanical simulations. The work presented shows the details of the design process including the neutronics calculations using Monte Carlo radiation transport codes [GEANT4, MCNP] the CFD and mechanical calculations using the ANSYS software suite and the MHD calculations using analytical solutions and OpenFOAM calculations.

Commissioning of the KATANA Water Activation Loop at JSI TRIGA reactor: dose field measurements

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

Water, as a primary coolant, will play a crucial role in the performance of fusion reactors since it generates an ionising radiation field throughout the facility after irradiation and activation, necessitating improved shielding for instrumentation and personnel. To support ITER, the KATANA irradiation facility, which utilises a closed-water activation loop, was successfully licensed, constructed, and commissioned at the end of 2023 at the JSI TRIGA research reactor in Slovenia. KATANA serves as a well-defined and stable source of high-energy gamma rays (6 MeV - 7 MeV) and approximately 1 MeV neutrons. Such a high-energy irradiation facility will facilitate various experiments based on water activation. The ultimate objective of KATANA is to perform benchmark-quality experiments, such as the validation of fluid activation codes, and to establish itself as a reference facility for the calibration of high-energy gamma detectors, which will significantly support the operation of ITER and other future water-cooled fusion reactors.

During the commissioning phase, with a strong focus on safety, gamma and neutron dose rate measurements were carried out around the KATANA circuit (within the experimental area enclosed by concrete walls) and throughout the reactor hall to create a comprehensive dose rate map across various operational scenarios. These measurements were undertaken in three phases, concentrating on different radiation sources: (a) background levels from the reactor without activated water, (b) during steady-state reactor operation with activated water, and (c) with activated water during pulse-mode operation. Dose rates, expressed as dose rate equivalents of H*(10)/time, were measured using a certified neutron probe (Berthold LB 6411) and two gamma detectors: a pressurized ionisation chamber (Fluke Victoreen 451P-DE-SI-RYR) and a scintillator probe (Automess 6150AD-b/H). The peak gamma dose rate observed was up to 5 mSv/h in the immediate vicinity of the main observation area of the circuit (Snail head), with neutron contributions significantly lower, by more than three orders of magnitude. Due to these elevated dose rates, the experimental zone within the concrete walls has been designated as a red zone, subject to stringent access restrictions. Outside these walls, however, the dose rates remained below the limit of 10 µSv/h, which indicates that no additional shielding is required. The mapping of the dose field has provided important insights into the radiological safety of personnel and established guidelines for the optimal placement and arrangement of detectors within high gamma and neutron fields for future experiments at the KATANA facility.

JADE v4, a more robust and expandable architecture for neutronics V&V

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Please, indicate type of contribution:	Poster session	☐
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The development of JADE started in 2019 as a joint effort between Fusion For Energy, the University of Bologna and NIER engineering. At the time, JADE was focused on bringing automation and standardization to the Verification and Validation (V&V) procedures of nuclear data libraries. From the very beginning, the ambitious objective was to build an expandable open-source framework that could centralize the efforts of a community, the one of nuclear data libraries evaluators, that was characterized by limited and fragmented manpower. Since then, the tool was applied with success in helping the V&V procedures of many new nuclear data libraries releases, with a special focus on FENDL, by which JADE was officially endorsed. In the last couple of years, the tool development has been pushed forward by Fusion For Energy and UKAEA with a new objective: use the JADE framework also to perform code to code comparisons. This new collaboration led to JADE v3, a first proof of concept that ported the tool on the linux platform and allowed for the first time to run benchmarks not only with MCNP but also with OpenMC. During this phase, the JADE environment was restructured and components with different scopes were made independent and separated in different repositories. Despite all these achievements, it became clearer that adding new features to JADE was becoming increasingly harder because JADE was not initially conceived with this broader scope in mind. The code base reached a high degree of complexity and was becoming a barrier for the onboarding of new developers in the project. These reasons led to the work presented here, which focus on the latest iteration of JADE development: JADE v4. The entire core architecture has been refactored following two main principles. The first is that JADE is conceived now from the very beginning to be a framework for comparison of code-library results and not simply for library to library or for code to code. The second is that the csv data produced by JADE, which are essentially the results of the different simulations in a table format, will be now a key interface. That is, all post-processing that is transport code dependent will end with the production of the csv. This allows the JADE post-processing (plots and excel summaries) to be completely transport code independent. A side benefit of this is that a better interface is created for the JADE web app. Moreover, this will allow to expand the JADE benchmark suite only through configuration files and input templates, without the need for additional coding. Finally, the extensive refactoring also allowed to implement a series of software design best practices which significantly increased JADE robustness and expandability.

Layered Shielding Solutions for Spherical Tokamak Designs

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Please, indicate type of contribution:	*Poster session*	☐
	Oral Contribution	☒

The D-T Fusion process produces a high flux of 14 MeV neutrons, which, while being the primary source of energy production, damages structural and functional materials within the fusion device that has detrimental effects on the materials including the high temperature superconducting magnets, one of the key components of spherical tokamaks (ST). The compactness of the STs requires the use of highly efficient neutron shields. In order to safeguard magnets, materials with the good neutron scattering, moderation, and absorption are needed to attenuate neutrons. Tungsten, known for its high scattering cross section, and carbon, boron, and hydrogen, known for their high absorption and moderation cross section, are some of the key materials that show promising neutron attenuation, absorption, and moderation. In this work, layered configurations of shielding material consisting of tungsten, tungsten carbide and tungsten boride or boron carbide or hafnium hydride that provide good magnet lifetimes, which is an indicative of neutron shield performance matrix, have been optimised and discussed. The layered configurations use different tungsten-based materials, e.g., tungsten alloy, tungsten carbide, tungsten borides, boron carbide and hafnium hydride which are considered as a good neutron shield as a monolithic. However, by using them in a layered manner in a sequence of tungsten followed by tungsten carbide, followed by the tungsten boride or boron carbide or hafnium hydride, provides us with even better shielding properties, ensuring there is no degradation of TBR. These materials have different TRL levels and provide increasing magnet lifetime. Shielding configurations facilitate the sequential advancement of shielding based on the requirements, applications, and R&D budget of fusion devices. Alongside their highly efficient neutron attenuation properties, these shielding materials must also possess good thermo-mechanical properties, as shielding will require complex shape and sizes and neutrons interact with the constituents of shielding materials, generating heat within the shield due to nuclear interactions. High nuclear heating in shield materials raises the requirement of additional cooling channels, thus demanding a good thermal conductivity to remove the heat from centre column shield (CCS) and maintain it at the desired operating temperature range.

The presentation discusses the overall shielding solutions to the STs that includes neutron shielding materials, cooling channel, and structural materials. The presentation will cover the two aspects of the shielding solutions: nuclear responses, calculated with the Monte Carlo N-Particles (MCNP) and FISPACT codes, and thermomechanical responses. The nuclear response includes the magnet lifetime, gas production, dpa, and radioactive waste hazards for layered shield configuration. Thermomechanical responses include the development of FEA methods including CAD designs of a ST with the structural materials and cooling channels, thermal and structural analysis of the proposed shielding solutions. However, due to the limited availability of thermo-mechanical property data for certain grades at operating temperatures, the presentation also discusses the key requirements for further developments.

Neutron transport simulations for an automated parametric integrated stellarator design.

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Please, indicate type of contribution: Poster session ☐ Oral contribution ☒

Providing neutronics analysis to support an automated, simulation-driven engineering design process requires modernizing existing neutronics toolchains. Proxima has embraced and is also defining modern design processes targeting the design of a Q > 1 stellarator and a power plant prototype stellarator, leveraging recent breakthroughs in the optimization of quasi-isodynamic (QI) stellarators.

The geometrically complex nature of stellarators lends itself to codified methods of geometry creation, whereas the rotational symmetrical geometry of a tokamak shifts the balance, making GUI-driven approaches relatively more feasible in tokamaks.

The production of automated, parametrically driven geometry from plasma and magnet configurations provides an excellent basis for neutronics analysis to be built upon shaping the entire workflow and promoting a modern workflow throughout.

The requirement to provide neutronics metrics (heating, DPA, dose, TBR, etc.) for a large number of designs necessitates a completely automated workflow, including the automated conversion of CAD-to-neutronics geometry conversion, simulation, post-processing, and integration into a software pipeline.

Towards this goal, Proxima actively contributes to nuclear data processing efforts, particle transport simulation software (OpenMC), geometry tools (CadQuery, OCP, DAGMC), meshing tools (cad-to-dagmc, assembly-mesh-plugin), and related aspects, including software packaging, distribution, testing, maintenance, documentation, and development.

In this presentation we showcase automated neutronics simulations carried out on geometrically distinct stellarator designs. We utilised hybrid Embree-vectorized DAGMC geometry for the device and CSG geometry for the bioshield This hybrid approach allows for accurately representing the highly curved nature of the stellarator while also maintaining efficient particle transport by combining the relative strengths of different neutronics geometry approaches.

This comprehensive, full-stack approach to contributing to open-source neutronics tools results in a highly deployable, reproducible workflow that is possible to integrate with the overall design workflow. Ongoing engagement in the ongoing effort to improve neutronics tooling is envisioned to enable a deployable, reproducible, and efficient neutronics workflow.

A ROADMAP PROPOSAL FOR EXPANDING USAGE AND IMPLEMENTATION OF OPENMC IN ITER

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

Nowadays, the ITER fusion neutronics community relies almost exclusively on MCNP (and its different patches such as D1S-UNED, Advanced D1S, DAG-MCNP, etc.) for conducting radiation transport calculations in support of nuclear analysis. In recent years, the number of OpenMC users and proponents is substantially increasing, as evidenced by the many presentations given during the 2024 ITER neutronics meeting. The transport code features diverse capabilities for geometry representation (such as CSG, DAGMC, and unstructured mesh (UM)), computational advantages (including transport and activation), dedicated API and potential for exploiting exascale compute with recent GPU portability developments. As an open-source code there are no restrictions for deployment on cloud and cluster environments. Therefore, OpenMC is a valuable candidate for international and complex fusion projects like ITER and a possible complementary tool to MCNP being useful also for code-to-code comparison.
For this reason, this contribution would like to present a possible roadmap for the increased usage and implementation of OpenMC in ITER, the largest fusion project under-construction, providing a detailed plan for enhancing capabilities and improving users engagement.
The aim of the roadmap is to identify the features of OpenMC that need further development for ITER needs, provide an overview of the formal process for using it in ITER final design activities (or QC1), and compile various global verification and validation (V&V) efforts specific to fusion applications.

This contribution does not represent any commitment by ITER or any other parties to adhere to the agreements or recommendations made here, being intended only to enhance coordination and as a foundation for future collaborations in this area.

Simulation of water activation in the KATANA activation loop of the TRIGA reactor

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

Water is one of the most commonly used coolants, it is used in both fission and fusion reactors. One of the peculiarities of using water as a coolant for components where there is a significant flux of neutrons is the activation of this water. As a result, the cooling circuit can become a significant source of radiation, affecting the design of reactors. In order to design and license the ITER fusion reactor, methods were developed to simulate activated water as a source of radiation, and at the same time, the need for experiments where these tools could be tested became apparent.

In 2024, the KATANA activation loop, aimed at improving the understanding of water activation processes, started operating at the TRIGA Mark II reactor at the Institute Jožef Stefan. One of the main applications of this loop will be the validation and optimization of computer programs used to calculate water activation and simulate activated water as a radiation source.

This study presents an analysis of water activation in the KATANA loop using the FLUNED tool, which integrates fluid dynamics simulations with neutron and gamma-ray transport for activation calculations. In this analysis, the particle transport was simulated with the Monte Carlo stochastic particle transport program MCNP, while water velocity fields were computed using the computational fluid dynamics program OpenFOAM. The activations of the oxygen isotopes ^{16}O, ^{17}O and ^{18}O and the decay of their activation products were taken into account: ^{16}N, ^{17}N and ^{19}O. For each isotope, the saturation activity was estimated at different flow rates and the dose fields were calculated. At a power of 250 kW, the maximum saturation activity values achieved were 7.1×10^7 Bq for ^{16}N at 0.5 l/s, 1.2×10^4 Bq for ^{17}N at 0.8 l/s and 1.1×10^7 Bq for ^{19}O at 0.1 l/s. These activities correspond to gamma ray dose rates of 5.7 mSv/h for ^{16}N and 0.31 mSv/h for ^{19}O, and a neutron dose rate of 26 µSv/h for ^{17}N, all calculated inside the instrumentation channel of the outer observation part.

Signal-to-background ratio calculations for ITER VNC

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

Vertical Neutron Camera (VNC) is an ITER diagnostic aimed at the reconstruction of the time- and space-resolved neutron and α-source profile in the poloidal plasma section as well as the ion temperature profile. It is composed of 11 neutron collimators, each having 1 detection unit, consisting of 2 fission chambers and 2 diamond detectors. Upper VNC is located within upper port #18 and contains 6 lines of sight, lower VNC is located within lower port #14 and contains 5 lines of sight.

High flux density of background neutrons is a serious problem for the VNC. Background neutrons can be split into two terms. The first one is indirect neutrons, i.e neutrons that entered the detectors not through the collimators (either by flying through the VNC structure uncollided or by scattering on the surrounding materials). The second term is backscattered neutrons, i.e. neutrons that scattered on the elements of the blanket of divertor opposite to the collimator that enter the collimator and reach the detector. ITER upper port-plug has very limited space, so increasing collimators length is impossible, so during optimization of VNC design [2], collimator shapes were broadened in toroidal direction. That decision increased number of direct neutrons, yet further broadening is impossible due to growing fraction of backscattered neutrons.

We calculated indirect background component by filling collimators with surrounding materials. Neutron calculations were done in ITER C-Model [3]. Indirect neutron flux calculations were done in a model with collimators filled by surrounding materials. Backscattered neutrons are estimated by flagging neutrons that passed through the cells, located opposite to the examining collimator. Direct neutron flux is estimated as a difference of total and background neutron fluxes. Additional calculations were done to analyze importance of neutrons, which underwent small-angle scattering in collimator channels, to the direct count rate. Neutron energy spectra and diamond detector PHS was calculated as well. It is shown that high energy threshold allows for much better signal-to-background ratio.

The work is supported with the state contract of 20.03.2024 № H.4a.241.19.24.1024 «Special equipment development, experimental manufacturing, testing and delivery preparation to meet Russian Federation's liabilities on ITER project in 2024 year».

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Discontinuous-Galerkin based Deterministic Neutronics for Stellarator design

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	Oral Contribution x☐

Stellarators are fusion power plant candidates with complex 3D geometry (and associated strongly heterogeneous neutron loads, both poloidally and toroidally), inherent steady-state capabilities and no current-induced disruptions. Compared to the tokamak alternative, stellarators are thus more difficult to build and design but significantly easier to operate. Although this 3D shaping complicates design, it also allows for the optimization of the geometry for given parameters such as cost or size. Optimizing the geometry does require a fast design cycle, including quickly assessing relevant engineering constraints.

However, engineering models for stellarators compatible with such a fast design cycle have only been introduced recently [1], and a blanket model that can quickly assess the neutronic viability of the design (e.g. tritium breeding ratio, fast neutron flux at the coils) is still lacking.

In principle, conventional Monte-Carlo methods could be used to evaluate the neutron response, but they can, especially in the 3D curved stellarator geometry, require significant computational resources and manual work, both in conflict with the desired fast design cycle. Furthermore, reduced models as developed for tokamak system-codes are incapable of modelling the highly heterogeneous neutron load in stellarators.

We have been developing a deterministic method for efficiently solving the neutron transport equation in (possibly parametric) stellarator geometry, providing a middle ground between expensive Monte-Carlo methods and inaccurate reduced models. It uses a discrete ordinates (S_N), multigroup velocity space discretisation with an unstructured mesh, arbitrary order, discontinuous Galerkin spatial discretisation. The discontinuous Galerkin formulation allows for an efficient transport 'sweep', limiting memory requirements and improving computational speed. Modern matrix-free iterative solvers are used to solve the within-group equation, limiting the number of sweeps required significantly. The model can be used in 1D, 2D, and 3D.

Benchmarks against both analytical solutions and Monte-Carlo methods are shown in slab geometry, including a three-dimensional breeding blanket test relevant to fusion. Results very close to the reference solutions are obtained in significantly less computational time. Preliminary results in stellarator geometry will also be shown.

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Development of a rapid method for assessing the impact of material impurity deviations on the shutdown dose rate

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The shutdown dose rate is a critical quantity evaluated in the nuclear analysis of ITER. Calculations of this nuclear response through the tokamak complex require computationally expensive simulations using advanced radiation transport methods and workflows, often spanning several weeks. The shutdown dose rate depends directly on the chemical composition of materials, not only alloying elements but also, and in many cases more significantly, their impurity content. As a result, ITER has project requirements stipulating a maximum permissible content of these elements. A growing number of non-conformities in relation to these requirements for manufactured components are being raised in parallel with the advancing construction of ITER. Conducting a detailed assessment of the impact of the deviation in impurity content for each instance of non-conformance poses a major risk to the ITER baseline given the critical input of nuclear analysis. Here, a new tool, F4Epurity, is presented to allow a rapid estimation of the resulting change in shutdown dose rate. Using available radiation maps for the tokamak complex and compatible cross-section data, the reaction rates and subsequent activity is derived analytically for a given irradiation scenario and decay time. All important reaction channels are considered, fully accounting for metastable states, with decay chains traced to the point of a stable daughter product. Pre-computed nuclide specific dose factors are then used to determine a dose in Sv/hour. A 3D map of the dose is derived based on analytical expressions for currently supported point and line source geometries in the format of a VTK file. The code also provides a native plotting functionality to display 2D slices superimposed with the geometry. Additional capabilities include a shielding estimator, option to export MCNP format source definition and support for input of pre-computed activities. We also present initial efforts investigating the use of ADVANTG to pre-compute maps of the adjoint flux. These can be used to determine the change in a given tally's dose response resulting from a deviation anywhere within the map, without the need for further calculations. The major advantage of this approach is in overcoming the largest approximation in the tool of unshielded and unscattered conditions. Examples of recent applications of the tool will be presented demonstrating its capability for providing input to safety assessments at ITER.

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EUROfusion Neutronics R&D Supporting ITER Nuclear Operations

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

Within the EUROfusion Work-Package Preparation of ITER Operations (WP PrIO), the sub-project "Neutronics, Nuclear Waste, and Safety" provides significant advancements in fusion neutronics through extensive research, development and experiments.

Key nuclear fusion relevant data and insights were gained from recent Deuterium-Tritium (DT) campaigns at the JET tokamak, related to material activation characterisation, functional material degradation, 14 MeV neutron diagnostics calibration, Test Blanket Module (TBM) detector testing, and Tritium Breeding Ratio (TBR) assessment. Benchmark experiments on neutron streaming and shutdown dose rates validated computational tools and nuclear data. Unique studies investigated water activation in JET cooling loops and Single Events Effects (SEE) on electronics during DT plasma operations.

In parallel, further neutronics activities were performed in close collaboration with F4E and ITER Organization (IO), specifically focused on ITER priorities. These include the development and validation of multi-physics methodologies for fluid activation, Activated Corrosion Products (ACP), modelling neutron and gamma sources during the various ITER phases, coupling neutronics with virtual reality for ALARA-compliant maintenance, conversion from CAD to Monte Carlo geometry, SEE on electronics. EU facilities such as the Frascati Neutron Generator (FNG), JSI TRIGA MARK-II reactor, CSM-RINA corrosion loop, and WEST supported these efforts. The main milestones include commissioning the KATANA water activation loop at JSI and the successful execution of initial water activation experiments there, corrosion tests under ITER baking conditions, and ACP water loop design at FNG. The design of the GENeuSIS (General Experimental Neutron Systems Irradiation Station) assembly for testing SEE on electronics and diagnostics in ITER-like neutron conditions was also finalized.

This work provides an overview of the neutronics advancements within PrIO supporting ITER licensing, nuclear operations, and their implications for DEMO design and future fusion technologies.

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Shutdown Dose Rate Assessments due to Water Contamination in EU-DEMO Primary Heat Transport Systems

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Water cooling systems in nuclear applications inevitably present material activation that can contaminate components which are not directly irradiated by the neutron flux. Activated Corrosion Products (ACP), in the form of deposits, ions, and cruds, can be transported out of vessel generating significant gamma fields. As a consequence, in routine hands-on maintenance operations, the working personnel could be then exposed to a relevant radiological hazard.
ACP have impact not only on Occupational Radiation Exposure (ORE), but also on waste management, machine availability, components qualification, and accident consequences determination. Their accurate estimation and the evaluation of the hazards associated are of utmost importance for a thorough safety analysis.
To provide sufficiently accurate data for ORE assessments, it is advisable to conceive a methodology to estimate precisely the source terms and then perform gamma transport calculations to produce the dose rates.
In this work, the shutdown dose rate maps are calculated for the EU-DEMO Divertor Primary Heat Transport System due to water contamination.
To do so, a three-steps calculation scheme has been put in place. Once neutron activation reaction rates are estimated, the total inventory of ACP is calculated using OSCAR-Fusion v1.4a. Then, a specific patched version of MCNP has been used to define gamma sources accordingly and to perform transport calculations to produce the dose rate maps.
Parametric results are provided for different design assumptions and operational regimes at the beginning of both short and long-term maintenances.
The definitive results will be of crucial importance to estimate the precise amount of collective dose, one of the most relevant safety objectives in the future fusion power plants licensing.

Return of experience on compliance and typical levels of impurities of relevance for radioprotection in ITER materials

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Materials under neutron irradiation undergo nuclear transmutation, a.k.a. activation, and become radioactive. Activation levels determine other nuclear responses such as shutdown dose rate (SDDR), contamination and decay heat. As a consequence, activation has an impact on several nuclear safety aspects such as occupational radiation exposure (ORE), accident consequences, releases and radwaste. Activation levels depend significantly on chemical composition, and not only of the main alloying elements but also, in many cases, of certain impurities. Accordingly, these impurities have an impact on the above nuclear safety aspects as well.

In order to ensure compliance with regulatory commitments related to the above nuclear safety aspects, and in line with existing practices in the nuclear industry, most ITER SRDs and PAs contain requirements related to impurities of relevance to radioprotection in chemical compositions of materials exposed to neutron irradiation, namely Co, Ta, Nb and Ni. Compliance with these requirements is typically achieved and verified by the arrangement of dedicated procurements with specialised suppliers, which produce specific and certified batches so as to guarantee conformity. The typical ITER materials concerned with these requirements are the following:

- Aluminium alloys.
- Carbon steels.
- Copper alloys (e.g. Cu, CuCrZr, Al-Brz).
- Austenitic stainless steels (e.g. grades 304, 316, XM-19, 660).
- Nickel alloys (e.g. grades 625, 718).

Moreover, detailed elemental compositions for these materials become an essential input element of the nuclear analysis models used for determining radiation conditions throughout the facility and for safety demonstration.

Here we report the currently available results of ongoing efforts to gather relevant return of experience (RoX) on compliance with requirements and typical levels of impurities of relevance to radioprotection in ITER materials. This is done by painstaking collection and analysis of certificates of both purpose-built and off-the-shelf material batches. This information is intended to support the preparation of nuclear analysis models used for safety demonstration in a best-estimate rather than conservative fashion, as well as the impact assessment of deviations and non-conformities.

Neutronics Tools used at ITER, a general overview

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

ITER is a complex nuclear machine and is poised to become one of the most intricate radiation sources ever built, with its diverse spatial distribution, time dependency, and the presence of multiple energetic radiation types such as neutrons, gamma rays, and even electrons. Modelling these radiation fields requires a robust suite of tools that address various challenges encountered in different stages of the calculation process.

The primary code employed in ITER neutronics analyses is D1SUNED, developed by UNED colleagues, which is an MCNP patch capable of simulating ITER's different operational modes: mode 0: plasma operation, mode 1: hands-on maintenance, and mode 2: remote handling maintenance. Complementing this are SpaceClaim, used for model preparation and geometry generation, and ParaView, typically employed for result visualization and post processing.

To support and complement these main codes, many specialized tools have been developed and integrated into the neutronics workflows. These tools span a variety of applications, including:

Model Preparation and Conversion, Variance Reduction Tools, Model Integration, Post-Processing Utilities.

For isotope inventory and activation calculations, various codes have been utilized in the past. However, starting in 2025, FISPACT-II will become accessible for ITER analyses, with availability expected later in the year. This inclusion is anticipated to standardize activation and decay calculations across ITER projects, ensuring consistency in results and enabling compatibility with existing workflows.

These tools, together with the main codes, form a cohesive ecosystem for tackling the complexities of ITER radiation analysis. Emphasis is placed on streamlined workflows, reducing turnaround times, and maintaining high accuracy in results. Ongoing development efforts focus on further automation, improving user accessibility, and ensuring adaptability to ITER's evolving design needs.

This presentation will highlight the interaction between these tools, demonstrate examples from recent neutronics analyses, and discuss plans for future developments in the toolset to address challenges in ITER radiation modelling.

Point-kernel-based 3D ray tracing gamma dose rate calculator

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Please, indicate type of contribution:	Poster session	☒
	Oral Contribution	☐

ITER radiation sources are inherently complex due to their diverse spatial distribution, time dependency, and the presence of multiple radiation types such as neutrons and gamma rays. The standard approach to modeling these sources typically involves the development of MCNP-based models, which, while robust and accurate, often demand significant time for model preparation, simulation, and post-processing. This extended turnover time can present challenges in meeting the fast-paced requirements of iterative design processes. Furthermore, the lack of alternative tools makes cross-checking MCNP results with independent methods difficult.

To address this gap, a point-kernel-based 3D ray tracing gamma radiation dose estimator code is under development. Designed to provide faster and more flexible calculations for certain gamma sources, such as the ^{16}N water sources of the TCWS and the ACP sources, this tool offers an efficient alternative methodology. The code supports 3D geometry files in STL format and performs detailed ray tracing to evaluate attenuation from pre-defined source points to target points, outputting gamma dose rates.

Key features include a user-friendly GUI for setup, input/output management, and the ability to save and load input data for external editing. Users can specify gamma-line-specific parameters, such as dose conversion factors, mass attenuation coefficients, and build-up factors, on a material-by-material basis. Materials are represented as individual STL files, simplifying geometry preparation and supporting efficient, iterative design evaluations.

Multi-processing capabilities further enhance the tool's computational efficiency, making it suitable for large-scale problems with numerous source and target points. This makes the code an ideal candidate for design support tasks where quick turnaround times and flexibility are essential. Additionally, the tool complements MCNP-based workflows by offering a rapid and independent means of validating gamma dose rate results, thereby enhancing confidence in the overall analysis.

ITER Radiation Maps Status, Roadmap, and Plan for 2025

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This presentation aims to provide a comprehensive status update on radiation maps, which serve multiple critical functions, including safety assessments, design optimization, and qualification processes. It will outline the procedures currently in place, present the latest developments, and discuss the proposed approach to meeting various regulatory and design requirements. Given the fundamental importance of safety, these procedures undergo periodic reviews to ensure that risks are effectively mitigated. A recent comprehensive assessment has led to the development of a nuclear analysis roadmap extending through 2029. This presentation will detail the key findings of this review and the corresponding strategic measures outlined in the roadmap. To support upcoming RPrS update, the project has proposed an intermediate step: the production of the Mode-0 radiation map by the end of 2025. The detailed plan for this initiative will be presented, including its implementation strategy and expected outcomes.

Investigating the relationship between CAD model complexity and performance trade-offs for fusion neutronics

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	Oral Contribution	☒

The goal of this research is to predict how CAD model complexity influences simulation outcomes, supporting the development of an automated defeaturing tool for fusion neutronics. Understanding this relationship is crucial, as the complexity of a 3D model affects the accuracy of neutronics simulations and, thus, the reliability of the measured quantity.

In computational solid geometry (CSG)-based neutronics simulations, an overly complex CAD model can impact not only the final simulation outcome but also the initial conversion process from its original representation (often in STEP format) to the code's native CSG geometry. A model that is too complex may require defeaturing or simplification to allow for timely simulations or for the simulations to even run at all. However, it is difficult to agree on a precise quantification of model complexity: these may be assessed using specific geometric characteristics of the model itself (e.g., number of faces, vertices) or through subjective perception of model complexity.

Some recent work has examined the connection between the perceived complexity according to CAD practitioners and certain quantifiable geometric and graph-based metrics. These resulted in the definition of metrics which can serve as sufficient approximation for model complexity. These metrics include the graph-based cyclomatic complexity and the number of node dependencies, whereas geometric metrics include the number of faces and vertices.

Here, we develop a workflow which evaluates such metrics and performs Monte-Carlo neutronics calculations, using a large dataset of CAD geometry (which includes over 3000 detailed and defeatured CAD models) that has been converted into CSG. Each data point of the dataset has been used as input for fusion-relevant OpenMC neutronics simulations. We then examine the relationship between geometry complexity metrics, runtime, and neutron flux tallies. Beyond the immediate research goal, we hope this research can benefit developers and those seeking to optimize CAD-to-CSG workflows.

High-Resolution Neutron Spectrometer for ITER tokamak – design optimization using numerical methods.

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Please, indicate type of contribution:	Poster session	☐
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The High Resolution Neutron Spectrometer (HRNS) is one of the fundamental tools of the ITER plasma diagnostics, whose operational role is the neutron measurement of the nT/nD ratio in the plasma core. The main objective of this work is to perform Monte Carlo (MCNP) calculations for the HRNS system, with special attention to neutron transport, shielding efficiency and the influence of surrounding components on the accuracy of the spectrometric measurement.
The development of nuclear analyses at the preliminary design level is crucial to define the performance of the HRNS system according to its requirements. IFJ PAN conducts nuclear calculations and analyses in support of the HRNS system, ensuring their compatibility with the development work in the PDR phase. These calculations include neutron and gamma-ray transport simulations based on the current system geometry, taking into account all significant components.

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

Surrogate-Enhanced Simulation Workflows for Fast and Efficient Neutronics Calculations

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Please, indicate type of contribution:	Poster session	☒
	Oral Contribution	☒

Neutronics simulations are usually computationally complex, and many runs may be required to explore design space, validate component interactions, and optimise performance. However, these typically run in isolation, with results often being set aside after use, causing underutilisation of the computational resource and the data produced. Furthermore, simulations are usually time-consuming, whereas near-instant results allow scientists and engineers to make informed decisions faster. Although some attempts have been made to create surrogates emulating simulations, these often necessitate costly and inefficient initial training datasets.

We propose an automated approach to integrate an emulation layer around neutronics workflows, enabling building a near real-time predictive cache. When neutronics simulations are run via workflows, they iteratively train the emulation layer, allowing future rapid predictions with quantified confidence levels. Users can define acceptable uncertainty thresholds that determine if a prediction by the surrogate layer is sufficient; if so, a result is taken from the model resulting in fast feedback. If not, a full-fidelity simulation is run, and the results are both returned to the user and dynamically update the surrogate layer, improving the model's accuracy and confidence and thus enhancing future predictions.

This approach uses digiLab's Uncertainty Engine™, which employs uncertainty quantified machine learning models including Gaussian Processes to produce the surrogate models, as a wrapper around the Galaxy Workflow engine. The workflow engine employs standard software packages and processing techniques, set up as tools, which can then be (re)configured into scalable, actionable workflows that are FAIR, findable, accessible, interoperable and reusable. In this specific use case, we integrate parametric CAD modelling via Paramak and neutron transport simulations using OpenMC to predict the Tritium Breeding Ratio. Comparisons are made over a series of simulation runs to evaluate the time savings achieved by employing the surrogate layer at differing confidence levels.

This approach applies to neutronics, wider fusion energy design, and other applications where many expensive simulations are run. By reducing high-fidelity computations and enabling faster decision-making, the approach enhances efficiency while decreasing computational overhead, making it a valuable tool for scientists and engineers across disciplines.

Neutronic modeling and nuclear analyses for the DONES neutron source needs at IFJ PAN

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

IFMIF-DONES will be a powerful neutron source with a neutron spectrum similar to the one expected in future fusion reactors like DEMO. The base to produce the fusion-like neutrons is a nuclear stripping reaction produced by a deuteron beam impinging and penetrating a liquid lithium curtain flowing from top to bottom. As a result of this reaction the maximum neutron fluxes expected in DONES are up to 10^{14}n/cm^2/s.

Monte Carlo simulations are crucial in optimizing the design of the particular DONES modules. At IFJ PAN, numerical simulations using MCNP6+McDelicius code are used for a wide spectrum of DONES needs. The optimization of both the STUMM (Start Up Monitoring Module) design and Complementary Experimental Hall (CEH) shielding should be mentioned here.

STUMM will be the first neutron and gamma diagnostics system installed just behind the neutron source, so numerical modeling and nuclear analyses are crucial for properly optimizing the system to fulfill its requirements.

Shielding optimization of the Complementary Experimental Hall (CEH) is another important issue related to safety regulation. In this case, due to the complexity of the problem and poor particle statistics, variance reduction methods were needed. ADVANTG code was used for this reason. Calculations including simulations of neutron and gamma-ray transport based on the current geometry and all system components important in this matter are presented in this paper.

Acknowledgments

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We gratefully acknowledge Polish high-performance computing infrastructure PLGrid (HPC Centers: ACK Cyfronet AGH) for providing computer facilities and support within computational grant no. PLG/2024/017736.

FISPACT calculations for the development of a neutron activation system for STUMM

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Please, indicate type of contribution:	Poster session	☒
	Oral Contribution	☐

STUMM (Start-Up Monitoring Module) will be used in the commissioning phase of the IFMIF-DONES facility. It will be placed in the Test Cell in the same position as the HFTM (High Flux Test Module) assembly. The main mission of STUMM is to verify neutronic calculations, characterize neutron sources, and determine radiation conditions in the HF region. To fulfil these tasks STUMM will host many detection systems such as self-powered neutron detectors (SPND), micro fission chambers with U-235 and U-238 (MFC235, MFC238), ionization chambers (IC), gamma thermometers (GT) and a rabbit system (RS) filled with activation detectors. The considered design of the RS is based on a commercial off-the-shelf (COTS) system used in nuclear reactors where metallic balls move through a pneumatic system. The measurements of the activity induced in the balls will be used to determine the local neutron spectrum and validate neutronics models.

This work presents the results of FISPACT-II calculations performed to select the proper materials from which the balls inside the rabbit system could be made. The input for these calculations was the neutron spectra calculated inside the RS for each of the five levels and across the eight rigs of the STUMM container. The key reactions that can be used in the deconvolution of the neutron spectrum are discussed, as well as a possible scheme for irradiation, measurement and cooling of the balls.

This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant Agreement No 101052200 — EUROfusion). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.
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We gratefully acknowledge Polish high-performance computing infrastructure PLGrid (HPC Centers: ACK Cyfronet AGH) for providing computer facilities and support within computational grant no. PLG/2024/017736

Breeding Blanket Neutronic Optimization for HELIAS reactor based on the Dual Coolant Lithium Lead concept.

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Please, indicate type of contribution:	Poster session	☐
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The growing interest in the Stellarator Fusion Power Plant concept as a possible alternative to Tokamaks within the EUROfusion community is driving an increasing effort to bring the Stellarator concept to technological maturity. New initiatives are underway to develop a Dual Coolant Lithium-Lead (DCLL) BB for a Helical-Axis Advanced Stellarator (HELIAS), as the DCLL concept presents high potentialities to answer the specific challenges posed by the HELIAS configuration. However, given the complex geometry of a stellarator and the limited space available between plasma and superconducting coils, innovative approaches are required.

The use of the ad-hoc prepared modelling tool HeliasGeom, combined with the GEOUNED code, enables the fast generation of layered neutronic models for a 72º HELIAS period to assess the feasibility of new radial configurations from a neutronic perspective, and an extensive analysis of Tritium Breeding Ratio (TBR) and radiation damage was performed through radiation transport simulations using the MCNP code. The progress made in integrated coils into the simplified HELIAS neutronic model has allowed preliminary shielding studies using tentative radial builds, focused on radiation damage, nuclear heating, and neutron fluence in detail, identifying critical areas of the coils that require further protection to prevent quenching.

The introduction of new modelling tools as ParaStell and DAGMC has allowed the generation of new layered neutronic models with variable thickness layers, which allows a better optimization of the thicknesses in the most critical areas of the model. Therefore, based on the aforementioned studies, some new optimized radial builds were proposed and analysed in this work, looking for a balance between TBR and coils shielding in the most critical areas.

Simulation Activities for the Calibration of ITER Neutron Diagnostics

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

During ITER operations, DD and DT fusion reactions will produce 2.45 MeV and 14.1 MeV neutrons. Measuring the neutron yield, neutron flux and neutron emission profiles is necessary for quantifying the total fusion power and the fusion power density. To achieve this, the ITER tokamak will be equipped with a set of neutron diagnostics, including a neutron activation system, neutron flux monitors and neutron cameras. These diagnostics require accurate calibration. The calibration factors are determined using Monte-Carlo neutron transport codes and validated through *in-situ* neutron calibration campaigns for both DD and DT operational phases.

The size and complex geometry of the ITER tokamak, along with uncertainties in the nuclear data, are a challenge for neutronic simulations, which can impact the accuracy of the neutron diagnostic calibration and, consequently, of the fusion power measurements. To address these challenges, Monte Carlo models of the Neutron Activation System and the Divertor Neutron Flux Monitors have been developed to study the uncertainties associated with the tokamak environment and local geometry in the vicinity of the detectors. In addition, a Total Monte Carlo workflow has been implemented to assess the impact of nuclear data uncertainties on the neutron measurements.

This presentation will provide an overview of the computational efforts for calibration of the ITER neutron diagnostics, highlighting key challenges and strategies for improving accuracy.

Neutronic benchmark and validation experiments at the Frascati Neutron Generator

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Please, indicate type of contribution:	Poster session	☒
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Based on a duoplasmatron ion source and a linear electrostatic accelerator in which up to 1 mA deuterium ion beam (D+) is accelerated up to 300 keV against a tritiated titanium target, the Frascati Neutron Generator (FNG) produces 14-MeV neutrons providing a neutron source strength up to 1.5×10^{11} s^{-1} in support of primary international fusion neutronics experiments. In line with its role in fusion, two experiments devoted to benchmark and validation of some computational methods, codes and nuclear data are foreseen at FNG in 2025, focused on activated corrosion products (ACPs), and shielding properties of concrete.

Concerning ACPs, inside the cooling loops of ITER, the water-wall interaction causes corrosion, erosion and subsequent release of the so-called corrosion products, which get activated under intense neutron irradiation. ACPs are a moving and intense radiation gamma source exiting the conventional radiation shielding of the tokamak through cooling loops and they are a concern in terms of radiation exposure of workers. The accurate and reliable modeling of the generation, transport and deposition of ACPs in cooling loops is fundamental to assess the radiation exposure, and the existing codes as OSCAR-Fusion (i.e., the reference code for ACPs for ITER) are validated mainly in fission plants, thus requiring validation experiments under fusion-relevant conditions to account for the significant differences w.r.t. fission plants, mainly on radiation field, structural materials and thermal-hydraulic characteristics of water flow in cooling loops. Among validation experiments, within the EUROfusion WP PrIO SP-5, a task aims at reproducing thermo-hydraulic (water temperature, velocity, Reynolds number), chemical (pH, oxygen content) and neutronic conditions relevant for ITER in a water circuit at a smaller scale. CPs will be activated under FNG neutrons and injected in the loop in the form of activated ions and solid particles in a dedicated section. Activity due to ACPs deposited will be measured in the dedicated section of the loop with spectrometers, while a sampling line is foreseen to measure activity concentration in water in a well-type spectrometer. Considering the importance of copper in fusion (CuCrZr alloy is the primary candidate material for the water-cooling system of the divertor in ITER), the first experimental campaign, foreseen in 2025, will be dedicated to such a material. Another essential material in fusion is concrete, and to cover the need of neutronics experiments under fusion-relevant environments to improve accuracy in modeling radiation transport for shielding design, some experimental campaigns with ordinary and heavy concrete are planned at FNG, within EUROfusion WP BB. The DEMO-relevant ordinary concrete shielding benchmark experiment is foreseen in 2025, with a mock-up consisting of thirty-six 50x50x5 cm^3 slabs of ordinary concrete, its feasibility being demonstrated by a thorough MCNP pre-analysis with JEFF and IRDFF-II data libraries, respectively for transport and dosimetry. Neutron flux and fluence will be measured inside the concrete blocks at different depths of penetration both with passive (activation foils) and some active detectors, among which, the promising radioluminescent optical fiber developed by Laboratoire Hubert Curien and earlier tested at FNG within RADNEXT network for industry and research.

The scope of the present work is to give an overview of the mentioned activities, mostly from an experimental point of view, to describe their status, achievements, and criticalities emerged.

JSIR2S code system overview

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Please, indicate type of contribution:	Poster session	☒
	Oral Contribution	☐

A key challenge in the design of fusion reactors is to ensure safety through accurate predictions of radiation fields and dose rates during operation and after shutdown. Shutdown dose rate (SDDR) calculations are important for safety during maintenance and decommissioning. SDDR predictions are also used to protect reactor components.

Such calculations have contributed significantly to the development of advanced shielding concepts and maintenance protocols, including the safety framework at ITER. The findings from the Joint European Torus (JET) facility have further refined the SDDR assessment methods relevant to ITER.

The JSIR2S code system based on the R2S method is currently being developed at the Jožef Stefan Institute (JSI) and combines calculations of the MCNP transport code with the FISPACT inventory code. It has been evaluated for fission applications based on measurements at the TRIGA research reactor, and in recent years attempts have been made to evaluate the code also for fusion applications.

In this contribution the latest work with the JSIR2S code is presented. SDDR calculations were performed on a simplified tokamak model with a given irradiation scenario for the operation of the tokamak to calculate the shutdown dose rate at different cooling times. For each cooling time, an analysis was performed to identify the parent isotopes that contribute the most to SDDR. In addition, an isotope analysis was carried out to identify the parent isotopes that contributed the most to the SDDR.

The model used for the calculations was a simplified model of a large tokamak, based on the JET tokamak. The total neutron flux and the SDDR were calculated at three different positions in 5 cm radius spheres. The model consists of the following materials: air, Inconel, iron, copper, Collemanite, steel, microthermal insulation, carbon and water. The model was irradiated with a DD plasma source similar to the one, present in the JET tokamak following a simplified irradiation scenario corresponding to the JET irradiation scenario from 1988 to 2016. The scenario is comprised of over 20 years of simplified irradiation history with more detailed history for the last 30 pulses. The last pulses have a neutron yield of 1.5×10^{15} with an irradiation time of 13 s.

The calculated SDDR at positions A, B and C are (4.87 ± 0.39) µSv/h, (3.44 ± 0.31) µSv/h and (0.74 ± 0.16) µSv/h 12 days after shutdown. At the cool-down time 120 days after shutdown, the SDDRs at positions A, B and C are (1.89 ± 0.15) µSv/h, (1.38 ± 0.11) µSv/h and (0.18 ± 0.05) µSv/h, respectively. The partial isotope analysis to determine the dominant parent isotopes revealed that isotopes such as Cu63, Cu65 and Mn55 contribute most to the SDDR 2 hours after shutdown, while isotopes like Ni58, Fe58 and Cr50 dominate the SDDR 120 days after shutdown.

Keywords: Shutdown dose rate, JSIR2S, isotope contribution

Radiation transport model development with Gitronics applied to JT-60SA

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

Gitronics is a novel, modular and Git-based approach for the management and development of radiation transport models being developed in Fusion for Energy. A pilot case study is undergoing for the development of an envelope-based MCNP model of the JT-60SA reactor. A private server hosted by Fusion for Energy is being used to host the pilot case. The benefits of the Gitronics methodology are evident and presented in this publication. The modularization of the inputs and separation of concerns have removed the challenges of working with a single monolithic input file as MCNP usually requires. The use of Git for version control and GitLab for enhanced collaboration and error tracking and project management is highlighted in this work. New features and systems to the reactor model are being added in a progressive way with the aid of Gitronics. These additions aim to improve the level of detail and accuracy of the model, and the assessments performed with it while being properly tracked and easily reviewed.

Monte Carlo simulation for optimization of a neutron spectrometer based on GEM detector.

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

A new neutron spectrometer concept based on a Gas Electron Multiplier (GEM) detector is presented. The proposed system is devoted to future spectrometry applications for fusion plasmas (ITER and beyond. The operating principle of the proposed spectrometer involves a collimated neutron beam interacting by elastic scattering with a thin foil of hydrogen-rich material like polyethylene, producing recoil protons. The spectrum of proton-deposited energy can then be correlated with the neutron energy spectrum coming from the plasma core.
The feasibility of the proposed idea was investigated using Monte Carlo simulations and then tested on the IGN14 neutron generator (IFJ PAN laboratory), based on the GEM demonstrator especially refurbished for this purpose. The geometry of the experimental set-up was optimized to estimate the expected proton and neutron radiation background in the measurement area as well as to calculate the proton spectrum coming from the converter to the detector active area. This work presents the results of this feasibility study based on theoretical modeling and computer simulations of a possible prototype detector configuration.

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Application of OpenMC for nuclear analysis at fusion facilities

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Please, indicate type of contribution:	Poster session	☒
	Oral Contribution	☐

The development of fusion energy as a sustainable power source requires accurate neutronic calculations to optimize facility design, ensure safety, and evaluate performance. This study explores the application of OpenMC, an open-source Monte Carlo particle transport code, for neutronic analysis in fusion facilities. OpenMC is a flexible and advanced tool that excels in modeling complex fusion systems, including tokamaks and stellarators.

This research focuses on key neutronic aspects relevant to fusion facilities, such as neutron flux distribution, tritium breeding ratio (TBR), nuclear heating, and radiation shielding. Using OpenMC, neutron and photon transport was modeled and benchmarked against MCNP in support of design efforts for fusion facilities like the EU DEMO and the beam-target tokamak-type Volumetric Neutron Source (VNS), incorporating detailed geometric representations of plasma-facing components, breeding blankets, and shielding components. Also, this work provides valuable experience and insights into routine neutronic workflows for fusion applications, addressing key challenges such as modeling repeated structures (lattices), the influence of plasma source form and distribution, and the application of weight windows for variance reduction.

The results highlight OpenMC's potential as an efficient tool for neutron transport calculations in fusion energy research, supporting the development and commercialization of fusion technology. Additionally, this study underscores the importance of accurate neutronic modeling and the role of open-source codes like OpenMC in addressing the challenges of fusion reactor design and optimization. These findings lay the foundation for future studies aimed at improving the performance and safety of fusion energy systems.

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An Overview of STEP Nuclear Analysis

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

This work has been funded by STEP, a major technology and infrastructure programme led by UK Industrial Fusion Solutions Ltd (UKIFS).

The Spherical Tokamak for Energy Production (STEP) is a UK programme to design and build a prototype fusion energy plant. STEP is an ambitious programme that will demonstrate the ability to generate net energy, fuel self-sufficiency and a route to commercialisation of nuclear fusion.

Relative to other concepts, a compact spherical tokamak presents significant additional neutronic design challenges, principally the shielding of superconducting magnets in the central column, sufficient tritium production, and high component heat loadings. Alternate design solutions are required to address these challenges and this paper describes the different design choices that have been made to meet reactor requirements.

Nuclear analysis plays a critical role in STEP and impacts the design of a large proportion of components within the bioshield and the reactor building. Flexible and robust analysis is required to drive the design but this must also be supported by equally robust analysis from other functions. This talk presents an overview of the reactor concept, design challenges, calculation methodology, and design choices for STEP. There is also discussion of the approach for managing design uncertainty, neutronics results, and calculation workflows.

Application of the TUD-W Benchmark for Nuclear Data and code Validation

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

Tungsten (W) plays a crucial role in fusion reactor designs, particularly as a plasma-facing material. The recent 2023 ITER rebaselining, which replaced the beryllium First Wall with tungsten, has further increased the demand for accurate tungsten nuclear data to ensure reliable neutronic simulations. This study presents a validation of tungsten nuclear data using the TUD-W SINBAD benchmark, recognized as the highest-quality benchmark for this purpose, but also for OpenMC software. Simulations were conducted using both MCNP6.2 and OpenMC transport codes to compute neutron and photon flux spectra at four depths (5, 15, 25, and 35 cm) within a thick tungsten alloy (DENSIMET) block irradiated by 14 MeV neutrons. Experimental data, measured with an NE 213 scintillation spectrometer, served as a reference for validation. Advanced statistical optimization techniques, including weight window generators and Python-based automation scripts, were applied to refine the simulations.

Comparisons of simulation results using various nuclear data libraries were performed with the JADE tool. While general agreement was observed, significant discrepancies arose above 6 MeV, with FENDL3.2c showing the best performance. OpenMC results, obtained through a benchmark conversion process, confirmed that the observed deviations were likely due to nuclear data rather than Monte Carlo code differences. Further analysis using MCNP's PKMT card identified high-energy photon discrepancies originating from (n,γ) reactions in W-183 and inelastic scattering in Fe-56 and Ni-58 impurities.

This study supported the validation of FENDL3.2c W nuclear data and endorsed the methodology for the conversion of the experiment into OpenMC format, proving also its additional value in the fusion community.

Automating neutronics and coupled multi-physics simulation for stellarator fusion pilot plants

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Please, indicate type of contribution:	Poster session	x
	Oral Contribution	☐

Engineering design of magnetic confinement fusion devices requires various calculations ensuring e.g. sufficient tritium breeding in the blanket (TBR), minimization of neutron heating to superconducting magnets, and building support structures for magnets. For stellarators, the design of the magnetic fusion core and engineering systems is more closely coupled compared to other magnetic fusion concepts like tokamaks. In particular, stellarator magnetic coils require careful consideration to ensure the design meets both engineering and physics requirements and constraints due to traditional modular coil shaping and requirements for proximity to the plasma. We have undertaken furthering the computational tools and workflows for these tasks. First, using the parametric CAD generator parastell [1], OpenMC neutronics simulations at the CAD-level [2] are performed to calculate TBR and nuclear heating of magnets. Variance reduction techniques are required for good statistics for the local nuclear heating in the magnets, and need to be further developed for better targeting of the regions of interest, considering the length scales of modular coils. Additional engineering analysis workflows will be needed to automate selection of magnetic coils at conceptual levels, including coupling with neutronics calculations. An isolated component of this has been automated optimization of structural element placement between modular coils using ANSYS structural optimization tools, to minimize coil deflection. Further coupled neutronics and thermal hydraulic simulations can help determine coil feasibility. These efforts to at minimum semi-automate the engineering aspects of stellarator design, to compliment and inform the plasma physics optimization can greatly improve upon the feasibility range of future nuclear stellarator designs.

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F4E workflow for nuclear analyses

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

The F4E neutronics team is embedded within the Engineering Analysis Unit of Fusion for Energy (F4E), Europe's Domestic Agency (EU-DA) in the ITER project. Their activities consist of performing nuclear analyses for both ITER and JT-60SA tokamak reactors, developing and maintaining tools used in fusion neutronics, and constant optimization and improvement of their workflow. This presentation will move between different steps of the state-of-the-art F4E workflow for nuclear analyses, describe recent updates to the developed tools used in those steps and provide examples of the latest nuclear analyses done by the F4E team or the suppliers.

The preprocessing of the CAD models is a tedious process, which can be mitigated by using the in-house developed collections of tools **RadModeling**, created to streamline the simplification of geometries by tracking the changes to every component independently. Translation of the geometry is done by using an open-source tool **GEOUNED**, developed by UNED and further improved with support of F4E. The resulting MCNP input decks can be integrated into the reference models with the use of **F4Enix**, a python package created for pre- and post-processing operations in the workflow for nuclear analyses. Configuration control of the existing reference model has shown to be significant challenge. For this reason, F4E developed **Gitronics**, a novel, modular and Git-based approach for the management and development of radiation transport models. Validating and verifying nuclear data libraries is crucial to provide reliable nuclear responses in radiation transport calculations, which can be performed using **JADE V&V** python package, recognized by IAEA as official V&V tool for FENDL.

While presenting different steps of the F4E workflow for nuclear analyses, various examples of recent F4E activities in the area of nuclear analyses for ITER and JT-60SA will be given. Furthermore, F4E contribution to EUROfusion's workpackage WPrIO will be highlighted, focused on the application of the RSTM tool to the closed water activation loop called KATANA, commissioned at the TRIGA Mark II research reactor of the Jožef Stefan Institute (JSI TRIGA) in Slovenia.

Nuclear data for radiation transport in fusion devices: The Fusion Evaluated Nuclear Data Library (FENDL)

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Please, indicate the type of contribution:	Poster session	☐
	Oral Contribution	☒

Nuclear data is fundamental to radiation transport simulations. The Fusion Evaluated Nuclear Data Library (FENDL) is a comprehensive and validated collection of cross-sections and energy-angular distributions essential for modeling particle transport in fusion devices. Since its inception in 1987, the FENDL project has been coordinated by the IAEA's Nuclear Data Section (NDS), with multiple versions released over the years. The latest version, FENDL-3.2c, includes evaluated nuclear data for incident neutrons, protons, and deuterons, continuing the tradition of previous releases.

A locally updated version of the NJOY-2016 processing system has been used to generate application libraries for both Monte Carlo and deterministic transport codes. The FENDL library has undergone rigorous verification and validation through a series of numerical and experimental benchmarks. One example of a verification and validation tool is the JADE system, which has been instrumental in ensuring data reliability.

To facilitate the maintenance, updating, and tracking of modifications to FENDL files, a dedicated data management system has been implemented. This system ensures full traceability of updates and allows end users to generate processed files directly on their computational infrastructures. Additionally, the packages *endf-parserpy* and *endf-userpy* have been developed to manage evaluated nuclear data files. These tools enable the manipulation, interpretation, and visualization of information in ENDF-6 and PENDF formats, making them useful for updating, verifying, tabulating, and analyzing FENDL data.

This contribution provides an overview of the status of the FENDL library, the results of its verification and validation process, and the methodologies and tools used for processing, validating, and managing the library. It also outlines ideas for the development of a new FENDL version incorporating covariance data, particularly considering recent releases of evaluated nuclear data libraries such as ENDF/B-VIII.1, JENDL-5, TENDL-2023, and the upcoming JEFF-4.

Nuclear Analyses supporting the ITER Radial Neutron Camera design development

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

The Radial Neutron Camera (RNC) is a key ITER diagnostic system, designed to detect uncollided 14 MeV and 2.5 MeV neutrons produced by deuterium-tritium (DT) and deuterium-deuterium (DD) fusion reactions. It consists of an array of detectors that span a complete poloidal plasma section along collimated Lines of Sight (LOS). The primary function of the RNC is to evaluate the neutron emissivity/α-source profile and the total neutron source strength, providing spatially resolved measurements essential for fusion power estimation, plasma control, and plasma physics studies.

The RNC layout consists of two fan-shaped collimating structures that view the plasma radially through vertical slots in the Diagnostic Shielding Module (DSM) of the ITER Equatorial Port 1 (EP01):

- the in-port sub-system, enclosed within a removable cassette inside the EPP1 diagnostic shielding module, dedicated to probe the plasma edges by means of 2 sets of 3 LOS lying on different toroidal planes;
- the ex-port sub-system, designed for plasma core coverage, consists of 16 interleaved LOS embedded in a massive shielding unit extending from the EP01 closure plate up to the Bioshield Plug through the Port Interspace.

In the frame of the ongoing RNC design development, supporting nuclear analyses are focused on the optimization of the diagnostic measurement performances, since they are assumed as a prioritized design driver. Additionally, nuclear load assessments have been conducted to evaluate the structural integrity of the RNC detectors and associated components under plasma operation conditions, providing guidelines for further design improvements. The most significant findings are presented herein.

Updates on nuclear analyses for ITER Diagnostics Equatorial Ports

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

Nuclear analyses are essential for the ITER port integration design and shielding optimization. In recent years, ENEA has carried out detailed nuclear assessments for three equatorial diagnostic ports at different stages of their design. These analyses have contributed to the Preliminary Design Review of Equatorial Port #2 and are currently ongoing for the Final Design Review (FDR). Calculations of nuclear quantities inside the Port Plug (PP) have been completed, while analyses in the Port Interspace (PI) are scheduled for late 2025.
Equatorial Port #8 (EP#8), which is also under FDR, presents unique challenges due to its proximity to the Neutral Beam Injection (NBI) sector. To address these challenges, analyses have been carried out using a 120° sector model extracted from E-lite, which includes the NBI sector. The calculations include the Shutdown Dose Rate (SDDR) in both the Port Interspace (PI) and Port Cell (PC), and include the breakdown of the components contributing to the dose rate in maintenance areas. Additionally, a sensitivity analysis has been conducted to assess the cross-talk between adjacent ports and the contribution to the SDDR of the gaps around the PP.
EP #12 has been chosen for these analyses as it is considered a representative diagnostic port. It passed FDR and is currently under the Manufacturing Readiness Review. The studies focus on modeling and calculations during and after plasma operations, considering areas from the Port Plug to the Port Interspace in the ITER 40° C-model, and the Port Cell structures, integrated into a sector of the tokamak complex model. These analyses include secondary sources for neutrons, prompt and decay photons. Additional sensitivity studies focus on the impact of impurity content in some components such as cooling pipes and PI beam structures. Assessment of nuclear quantities in different maintenance configurations is also ongoing.
All calculations have been performed using the D1SUNED v3.1.4 code. Neutron and gamma fluxes and nuclear heating are calculated during machine operation, radiation damage, gas and tritium production are calculated at the end of life, and SDDR is calculated 1 and 12 days after shutdown. The results of these analyses will be presented and discussed.

Simulations of the neutron field for streaming analyses in DT operations at JET

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

The Joint European Torus (JET) operated in 2023 with a deuterium-tritium plasma and neutronics experiments were performed for validating in a real fusion environment the neutronics codes and nuclear data applied in ITER nuclear analyses.

The JET torus is a complex machine and during the last four decades numerous smaller and larger components, predominantly experimental equipment, have been placed in the vicinity of the JET machine, forming a very complex environment. Due to the inhomogeneous geometry with larger number of ducts and streaming paths through which neutrons can leak from the torus and through the torus hall, it is very challenging to calculate the neutron field outside of the JET vacuum vessel.

The work on simulations of the neutron field at larger distances from the torus is presented. It is performed in order to be compared to measurements. In particular, the fluence of neutrons passing through the penetrations of the JET vacuum vessel and inside the torus hall was calculated in order to assess the capability of state-of-the-art numerical tools to correctly predict the radiation streaming in large and complex geometries.

The calculations are performed in a two-step process using the deterministic code ADVANTG to determine the variance reduction parameters and with MCNP for subsequent calculation of the neutron field with the Monte Carlo method. The main streaming paths are identified.

New Developments in Attila4MC® Related to Fusion Neutronics

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

This presentation and demonstration will provide an overview of new features and modules in Attila4MC, with the goal of streamlining CAD based neutronics workflows using the MCNP® unstructured mesh.

Recently, a new "real-world" CAD workflow was introduced into Attila4MC. This workflow enables users to work directly from dirty CAD assemblies with hundreds or thousands of parts, and with small, "transport insignificant" part interferences, mostly eliminating the need for time consuming CAD cleanup and simplification. An integral part of this workflow is the new Attila4MC-Cottonwood™ Variance Reduction module, which performs CADIS and FW-CADIS variance reduction on complex models through an automatic arbitrary mesh refinement (AMR) deterministic solver. Cottonwood automatically generates the deterministic computational grid and MCNP weight window grid definition, based on a user supplied target half value layer (HVL) multiple. Local refinement is performed automatically in high gradient regions, growing to coarser elements elsewhere, resulting in a rapid deterministic calculation and effective weight windows variance reduction. Cottonwood directly reads the MCNP input deck and abaqus® mesh file, making it compatible with any upstream mesh generation workflow. Cottonwood currently supports both graphical user interface (GUI) and command line workflows. The new Fornax™ activation module will also be presented. Fornax provides MCNP users with a GUI driven workflow for shutdown dose rate (SDDR) calculations using the rigorous two-step method (R2S), along with isotope inventory calculations.

SPARC neutron diagnostics: from design to calibration and fusion power measurements

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	Oral Contribution ☒

Commonwealth Fusion Systems (CFS) in partnership with world centers of excellence in fusion technology are preparing for a fusion breakeven (Q_{fus}>1) demonstration in its flagship DT tokamak SPARC. Fusion power (P_{fus}) measurement, on which this mission goal majorly relies, is primarily achieved by neutron yield rate (dY_n/dt) diagnostics. Neutron-based fusion yield monitoring has been applied to several DT fusion devices, including in the recent milestones at NIF (USA) and JET (UK). SPARC, currently under construction and assembly in Devens, MA, USA, will deploy four systems for neutron diagnostics to augment this method. The diagnostics comprise: neutron flux monitors (NFM), neutron activation system (FOIL), radial neutron camera (NCAM), and magnetic proton recoil spectrometer (NSPC). An in-situ calibration using a well-characterized compact DT neutron generator is under development to directly obtain experimental calibration factors for NFM and FOIL. NFM provides the time-resolved (Δt~10 ms) total dY_n/dt (n/s), alongside a confirmation from FOIL's integrated DT yield (n/pulse) measurement. A redundant and independent measure of total dY_n/dt (n/s) will be obtained by combining a high-resolution spectrum of the core from NSPC, and a spectrometric emissivity profile from NCAM. These two systems leverage a uniquely designed opening in the diagnostics port for direct plasma lines-of-sight and extensive pre-characterization of their detectors, which allow for an optical (raytracing) calibration factor estimation. The proposed methods for dY_n/dt and P_{fus} are supported by high-fidelity neutronics models of the SPARC facility. Multiple models and their validation procedures are under development, utilizing both Monte-Carlo and deterministic solvers. An overarching target of uncertainty <10% on P_{fus} drives the careful design testing, modeling, and operational preparations for the SPARC neutron diagnostics, as discussed here. The projected ranges of uncertainty budgets for the main methodological factors: diagnostics performance, calibration, and modeling, are also presented here with a brief description of the specific preparations at CFS and partner institutions for each of those.

Relevance of the IFMIF-DONES neutron source to irradiation parameters of the EU DEMO breeding blanket

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Planning the strategy of fusion energy deployment with the DEMO-type fusion power plant fleet, we must qualify structural and functional materials to withstand and carry on their functions under intensive neutron irradiation. For the EU DEMO Helium-Cooled Pebble Bed (HCPB) Breeding Blanket (BB) materials qualification and testing, we propose to use the Medium Flux Test Module (MFTM) of IFMIF-DONES (International Fusion Materials Irradiation Facility – DEMO Oriented NEutron Source). To reproduce the multi-effect phenomena in the breeder zone of the HCPB BB, a dedicated HCPB BB mockup called BLUME (BLanket fUnctional Materials irradiation modulE) has been developed. The HCPB design is based on the tritium breeder pin concept. The Advanced Ceramic Breeder (ACB) pebbles are used for tritium (T) breeding and titanium beryllide (TiBe12) for neutron multiplication. The ACB pebbles are composed of lithium orthosilicate and 35 mol% lithium meta-titanate. To prove the concept of using the DONES neutron source for testing the BB materials in the DEMO-relevant radiation environment, neutronics analysis has been performed with the McDeLicious-17 code to reproduce the interaction of accelerated to 40 MeV deuterons (d) with liquid lithium at the DONES target. The nuclear reactions at the target produce neutrons with an energy spectrum up to 55 MeV peaking at 14 MeV, corresponding to the DEMO first wall. Compiling McDeLicious-17 with the On-The-Fly (OTF) Global Variance Reduction (GVR) technique makes it possible to run deep-penetration radiation transport calculations through up to 6.9 m thick concrete shielding around the DONES target. Neutron and photon fluxes, neutron damage (DPA), and nuclear heating density (W/cc) have been analyzed for BLUME's structural and functional materials. Nuclear heating 3D mesh-tally distributions have been calculated for BLUME's three materials: ACB, titanium beryllide, and EUROFER to set the heat source in the ANSYS subsequent calculations for thermohydraulic and structural analyses. The presented results demonstrated the relevance of the DONES neutron source to the EU DEMO fusion reactor parameters. Compared with the DEMO HCPB BB T-production of 3.3 mg/day per pin, BLUME has a similar value (3.7 mg/day) using 7 pins surrounded by the lead reflector. The results were normalized on the d-current of 125 mA generated by the DONES one accelerator. Two accelerators will double neutron flux and all nuclear responses, including T-production in BLUME of DONES.

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Type of contribution:	Poster session	☐
	Oral Contribution	☒

Parametric Neutronics Study and Machine Learning-Based Tritium Breeding Ratio Prediction for the ARC Tokamak

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	X

The ARC tokamak, a high-field fusion power plant under development by Commonwealth Fusion Systems, plans to achieve tritium breeding using a FLiBe-based molten salt blanket. This work evaluates the sensitivity of the tritium breeding ratio (TBR) to various design parameters within the ARC design space, which directly impacts the viability of sustained tritium production in fusion power systems. Parametrically generated OpenMC [1] models from a CFS-internal tokamak design tool simulate fusion neutron transport and assess tritium-producing reaction rates. A broad parametric study was performed, varying design factors such as lithium-6 enrichment, vacuum vessel layer thicknesses, port dimensions, and material properties to quantify their impact on TBR.

To enable rapid TBR estimation without full Monte Carlo simulations, we developed TransformerTBRNet, a transformer-based neural network that accelerates design iterations by capturing the multivariate dependencies between design parameters and TBR. The dataset was divided into training, validation, and test sets in a 70:15:15 ratio. After training for 10,000 epochs using an L1 loss criterion, the model achieved a training loss of 0.0025 and a validation loss of 0.0037. On held-out design points, TransformerTBRNet predicted TBR with an average absolute error of roughly 0.37%, assuming a nominal TBR of order unity. This study highlights a novel methodological framework for systematically assessing TBR sensitivity and demonstrates the significant impact of design parameter variations.

[1] Paul K. Romano, Nicholas E. Horelik, Bryan R. Herman, Adam G. Nelson, Benoit Forget, and Kord Smith, "OpenMC: A State-of-the-Art Monte Carlo Code for Research and Development," Ann. Nucl. Energy, 82, 90–97 (2015).

Challenges and achievements in the IFMIF-DONES neutronics activities

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IFMIF-DONES (International Fusion Materials Irradiation Facility – Demo Oriented NEutron Source) is an accelerator-based neutron irradiation facility for the study and qualification of materials used for the design, licensing, and reliable operation of fusion reactors such as DEMO. Neutrons are serving as a critical tool for irradiating and testing material samples, while also posing significant radiation safety considerations. This talk will provide a comprehensive overview of neutronics studies and nuclear analyses, starting with the accelerator where neutron and gamma radiation emerges from deuteron losses and deposition, progressing to the target and test systems where most d-Li neutrons are produced and utilized, and then to the systems and area where the radiation is shielded. Key aspects analyzed include neutron-induced material damage and gas production, nuclear heating, radiation doses during both beam-on and beam-off conditions, activation of structural components and liquids (such as lithium and water), and the assessment and optimization of radiation shielding across the accelerator systems, target systems, lithium systems, and building and plant systems. Additionally, this talk will highlight recent advancements in neutronics simulation tools, evaluations of nuclear data, and experimental efforts, which provide essential cornerstones for nuclear analyses in DONES.

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

Implicit Stochastic Uncertainty Propagation Scheme for Two-Step Monte Carlo Simulations for Computational Neutronics Codes

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Abstract

In recent years, the analysis of nuclear responses using two-step Monte Carlo simulations has become increasingly important in computational neutronics. At present, the evaluation of stochastic uncertainties to assess statistical convergence remains a challenge. Existing methods, mainly developed for Rigorous-Two-Step methodologies, exhibit limitations in effectively propagating stochastic uncertainties from each Monte Carlo simulation to the final nuclear response.

In this framework, an innovative implicit stochastic uncertainty propagation scheme is developed. The aim of this implicit scheme is to evaluate the total statistical uncertainty in two-step Monte Carlo simulations, overcoming issues of existing approaches related to the calculation of the covariance matrix of the radiation field in the first Monte Carlo simulation. The implicit scheme propagates the stochastic uncertainty of the final nuclear response due to the first Monte Carlo simulation by defining a random variable with specific criteria. Meanwhile, the stochastic uncertainty due to the second Monte Carlo simulation is directly provided by the Monte Carlo code.

The implicit stochastic uncertainty propagation scheme has been implemented in SRC-UNED and R2SUNED, two computational neutronics codes that use the output of the first Monte Carlo simulation as input for the second Monte Carlo simulation. Verification of the methodology against the brute force method demonstrates that the results align with expected statistical values.

Keywords: stochastic uncertainty, two-step Monte Carlo simulation, implicit scheme, covariance matrix.

Speeding up complex MCNP simulations using neural network source biasing

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Please, indicate type of contribution:	*Poster session* ☒
	Oral Contribution ☐

Nuclear analysis of fusion facilities, particularly within the ITER framework, requires highly detailed modeling of complex geometries and radiation sources. Monte Carlo codes, such as MCNP, are used in this context due to their high precision and capability to deal with these cases. However, due to the computational cost associated with achieving convergence in MCNP simulations, Variance Reduction (VR) techniques are essential. In cases where only a small fraction of the source phase space contributes significantly to the tally, source biasing is a more effective VR strategy than variance reduction methods.

This paper presents a novel VR methodology based on neural network (NN)-driven source biasing. The method extracts particle histories that contribute most to the tally, using this data to train a neural network that predicts the probability of contribution based on position, direction, and energy. The trained NN is then employed to optimize the source sampling distribution, prioritizing particles that are more likely to contribute to the tally.

To validate the proposed approach, a benchmark simulation was developed using the decay gamma source of N-16, relevant to the ITER Tokamak Cooling Water System (TCWS). The study compares conventional MCNP simulations with the NN-based source biasing approach, demonstrating a substantial reduction in computational cost. The Figure of Merit (FoM) improved from 7.6 in the analog simulation with 10^{10} neutron histories to 119 using only 10^{7} neutron histories, highlighting a 15-fold efficiency gain while maintaining tally accuracy.

The results confirm that NN-based source biasing significantly enhances computational efficiency in MCNP simulations, particularly for problems involving radiation streaming through narrow penetrations. The promising outcomes suggest that this methodology can be extended to more complex ITER-relevant scenarios, motivating further development and integration into existing MCNP transport codes.

Nuclear analysis of the IFMIF-DONES commissioning and start-up phases.

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Please, indicate type of contribution:	Poster session	☐
	Oral Contribution	☒

Future fusion reactors such as DEMO will need to withstand very intense, high-energy neutron fields that can induce significant material degradation. The IFMIF-DONES facility (International Fusion Materials Irradiation Facility – DEMO Oriented Neutron Source) addresses this challenge by producing a neutron source through the interaction of a 125 mA, Continuous Wave, 40 MeV deuteron beam with a liquid lithium curtain. This neutron source can be used for the study of the materials exposed to extreme irradiation conditions such as those of the first wall. However, the deuteron-lithium interaction will also give rise to neutron and photon fields that can pose a potential radiation hazard.

During commissioning and start-up phases, the beam will be operated in a limited manner, just to allow the technical team to adjust the alignment of the different accelerator elements along the beamline. In these cases, the deuterons will be stopped in a copper cone inside the High-Power Beam Dump (HPBD), placed on a dedicated sideline of the accelerator. Even though the operation will be limited, the interaction of the beam with the HPBD will produce a significant radiation field. The expected neutron source in these situations is close to $3*10^{14}$ n/s. This neutron source is much lower than the one anticipated in the main source (the Test Cell) during nominal operation. However, the HPBD is not as heavily shielded as the Test Cell. Furthermore, during the beam commissioning once the Superconducting Radio Frequency (SRF) modules are installed the deuterons will be accelerated up to 40 MeV for the first time. The deuteron-copper interaction will produce neutrons with energies up to 45 MeV. The high energy neutrons have higher mean free paths that can impact the doses during the operation outside the confined areas. But these neutrons can also induce nuclear reactions with high energy threshold that can affect the doses during the operation through the activation of the cooling water, as well as the doses during maintenance periods.

This focuses on the nuclear analysis of the commissioning and start-up phases of the IFMIF-DONES accelerator. The proposed modifications to reduce doses both during operation and shutdown, and the key findings relevant to the analysis of future nominal operation are presented.

Photoneutron activation of the primary cooling circuit of the Hylife-II reactor

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Please, indicate type of contribution:	Poster session	☒
	Oral Contribution	☐

Inertial Fusion Energy (IFE) has recently experienced an increase in interest, especially after the very positive and encouraging fusion yield results obtained in NIF in 2022 and later on. Several private and public initiatives have flourished, with diverse technological approaches for IFE. The approach followed by the Xcimer Energy Corporation (XEC) makes extensive use of the previous conceptual reactor design Hylife-II developed in the 90's [1][2]. This design made use of a flibe thick-liquid-wall to provide protection to the first wall, and also to obtain enough tritium breeding ratio (TBR) in order to achieve fuel self-sufficiency. This molten-flibe flow also transports the produced energy into a heat exchanger to a secondary cooling circuit, where a standard power cycle can be implemented.
The molten flibe flow in Hylife-II becomes very active after receiving the fusion neutron flux inside the reactor chamber. Short-lived radioisotopes like N-16, F-20 and F-18 are extensively produced, creating very intense and energetic decay gamma fields around the coolant pipes all along the circuit.
Activated flibe cooling circuits present a phenomenon that does not happen in water circuits. The extensive presence of beryllium in the flibe results in photoneutron production all along the cooling pipes of the primary circuit, since the energy threshold for photonuclear reactions is well below the present gamma energies. This neutron production is small compared to the radioactive decay rates, and hardly contributes to the global TBR, but results in an unexpected neutron field all along the cooling circuit, and the activation of pipes and the heat exchanger. Photoneutron production happens also in fission reactor designs using flibe as a coolant [3], but it is especially intense in fusion reactors due to the neutron practical energy threshold to N-16 production.
There are currently no computational tools capable of computing this activation phenomenon and its residual doses in a single step. In this work a split-calculation methodology is presented for computing then cooling circuit activation and resulting decay doses, applied to the Hylife-II design.

References:

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[2] F. Ogando, et al., "Preliminary nuclear analysis of HYLIFE-III: A thick-liquid-wall chamber for inertial fusion energy", Fus Eng Des 202 (2024)
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Study of the D1SUNED transport subroutines for future optimizations in benefit of ITER nuclear analyses

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The nuclear analyses in support of ITER are MCNP-centric. The reference models of ITER built for these analyses have evolved over the last decade. Initially, they consisted of MCNP models of sectors of the reactor such as the C-lite, the C-model, or the NBI model. Later, E-lite was conceived as a 360º reactor model to assess the inaccuracies introduced by previous models regarding non-local quantities and the machine-escaping flux. In parallel, several MCNP models of the ITER Tokamak Complex were released. Coupling the models of the reactor and that of its site using an intermediate source did not provide enough robustness for the ITER safety case. To overcome this issue, the prototype of the first ITER integral representation was built: the ITER full model. This model represents a milestone in the evolution of MCNP models.

With this perspective, since 2022, the UNED team has been working on the design of an MCNP integral model of ITER. However, the computational complexity of these models in terms of the number of cells, surfaces and materials has been increasing, going hand in hand with an ever-growing computational demand. This problematic trend, rooted in the design of reference models, is leading ITER nuclear analyses toward a bottleneck, especially regarding its safety case. Improvements have been made in the widely used D1SUNED code in the past, particularly focused on the loading phase of the MCNP simulations. Nevertheless, the transport phase remains the most computationally demanding. Even with high-performance computing infrastructures, the use of future ITER integral representations seems challenging.

With this motivation, we studied the most time-consuming subroutines of the D1SUNED transport algorithm when applied to an ITER MCNP reference model. To this end, we used two versions of the C-model: one preserving its typical heterogeneity and level of detail, and another with large regions homogenized. The timing results for the heterogeneous model indicate that three geometry-related subroutines account for 75.45% of the neutron transport time. When photon transport is also considered in this model, accessing the secondary particle bank becomes the main computational burden, consuming 42.90% of the transport time. In contrast, the results for the homogenized model demonstrate that homogenization accelerates the transport phase of the simulations. However, this improvement comes at a potential cost: the distortion of the radiation fields forecast.