15th-ITER Neutronics meeting and Fusion Neutronics Workshop -2024

Report of Abstracts

https://indico.iter.org/e/117

Abstract ID : **3**

Integrated Fusion Neutronics Workflow for OpenMC, MCNP, and Shift

Content

A fusion reaction creates high-energy neutrons: in particular, the deuterium–tritium fusion reaction creates a neutron with 14.06 MeV energy. The neutrons generated from this reaction then go on to interact with materials in the reactor, resulting in a myriad of unique effects that can be measured in detailed radiation transport calculations. Important metrics relevant for fusion reactors are as follows: (1) heat deposition, (2) neutron damage, (3) material activation, and (4) tritium breeding ratio.

Because of the complex geometry of fusion reactors and the wide range of neutron energies and materials present, homogenization and discretization of energy and space can present significant challenges in modeling a fusion system. Monte Carlo (MC) particle transport codes model energy and spatial dimensions in a continuous manner, which is a benefit that outweighs the increase in computational demand compared to the use of discretized methods (i.e., deterministic transport solvers). Numerous MC particle transport codes are used for fusion neutronics, but this work focuses on three in particular: Monte Carlo N-Particle Transport Code (MCNP) [1], Shift [2], and OpenMC [3]. This paper introduces an automation framework to integrate the three codes by developing modules that streamline input generation and output postprocessing.

The automation is to simplify multi-code verification efforts and to provide different MC code options for users with varying access levels (MCNP and Shift are export controlled). This framework is demonstrated on a tokamak design with an immersion blanket, similar to the affordable, robust, compact (ARC) type design [4], and assumes that the model geometries are in a computer-aided design (CAD) format, not in constructive solid geometry (CSG).

The work has also been expanded to automate secondary calculations for fusion reactor design metrics, such as shutdown dose rate, waste material dose rate, and multiplhysics coupling. By providing a generic Python framework with standardized input structures, different transport and activation codes can be used in a modular fashion, which enables unit-level validation of complex calculations such as shutdown dose rate.

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Submitted by **BAE, Jin Whan** on **Friday, December 1, 2023**

Abstract ID : **4**

Suppression of Rhenium and Osmium Production in Tungsten by Selective Isotopic Enrichment

Content

Tungsten (W) based materials are primary candidates for the plasma facing components (PFCs) in fusion power stations. A side effect of the use of W in neutron irradiation environments such as fusion is the transmutation to Rhenium (Re) and Osmium (Os), which are known to degrade the mechanical and thermal properties. In this study, neutron transport and nuclear inventory simulations were used to investigate a strategy of selective isotopic enrichment and/or depletion of W isotopes to supress the formation of Re and Os in a representative first-wall monoblock design. It was found that Re and Os production can be reduced by up to three orders of magnitude compared to natural W by depleting W-186 isotope to 100%, as 186W(n,γ) 187W followed by beta decay is the main pathway for Re production. Two design methodologies were investigated that considered 1) an approach with two discrete, depleted in W-186 regions, with a range of depletion degree investigated and 2) a graded approach with layers of different levels of depletion to emulate a functionally graded material.

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Status: SUBMITTED

Submitted by **ANDERTON, Mark** on **Friday, December 8, 2023**

Abstract ID : **5**

Development of radiation sources based on CAD models for the nuclear analysis of IFMIF-DONES lithium loop

Content

IFMIF-DONES (International Fusion Materials Irradiation Facility – DEMO Oriented NEutron Source) will be a neutron source for irradiating materials to be used in future fusion power reactors such as DEMO. This facility will be built near Granada city, Spain. The facility is based on a deuteron beam impinging onto a liquid lithium jet to generate a neutron flux suitable for fusion material irradiation. In the lithium loop, lithium and Corrosion Products will get activated by deuterons and neutrons, and produce Be-7 and Activated Corrosion Products (ACP). These products will distribute along the lithium loop, both dissolved in Li and deposited locally in the cold section. The representation of these complex decay gamma radiation sources is essential to perform radiological safety studies.

CAD2CDGS (Computed-Aided Design to Common Decay Gamma Source) is a new tool to create decay gamma sources based on CAD models. Complex sources are defined specifying the nuclide concentration per unit of volume or the total amount of nuclide in each group of CAD solids. Using CAD models allows a user-friendly source definition. These volumetric sources, codified into CDGS (Common Decay Gamma Source) format, are intended to be used with extensions of MCNP radiation transport code (cR2S, MCR2S, R2SUNED, D1SUNED, etc.). The use of CDGS format has several advantages over the native MCNP SDEF source definition capability.

CAD2CDGS tool is programmed in Python 3, based on an Open-Source CAD engine (FreeCAD, Open CASCADE) and EUROfusion codes (cR2S European SDDR tool) and it is distributed through EUROfusion IDM platform (https://idm.euro-fusion.org/?uid=2Q3MDZ). A first version with source code, manual and examples has been release.

Finally, CAD2CDGS tool has been verified in order to check the tool correctness. Also, the tool applicability to IFMIF-DONES facility has been demonstrated with the radiological zoning analysis of the rooms surrounding the Lithium Loop Cell (which accommodates the primary heat removal loop) due to the decay gamma field created by Be-7 and ACPs.

Keywords: fusion neutronics, IFMIF-DONES, source definition, CAD, Activation, Activated Corrosion Product.

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Comments:

It is preferred an on-line presentation rather than an in-person presentation.

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Submitted by **GARCÍA BUENO, Juan** on **Thursday, December 14, 2023**

Abstract ID : **6**

Nuclear Data and Uncertainty Qualification for Nuclear Fusion Reactor Design

Content

To ensure the operability and safety of a nuclear fusion device, a thorough assessment of the nuclear radiation impact on all components will be required. The more sensitive components will require shielding, and the uncertainty in shielding design arising from nuclear data and radiation transport methodologies means that some margin must be retained to ensure that the operational program of the device is not impacted. However, because of competing constraints on shielding thickness such as electro-magnetic forces, space, magnetic field, cost, etc., it is important that these uncertainties are properly identified, quantified, and minimized.

Furthermore, with the advent of high temperature superconducting magnets and a variety of novel fusion power plant concepts, new materials are being deployed and new shielding requirements are being specified. Fusion power plants will also have to demonstrate their tritium breeding capability. Therefore, the quality of the nuclear data used in radiation transport, activation, and tritium breeding for several new elements must be evaluated and, in some cases, improved. Limits on precision of neutron measurements also lead to a need for increased design margin of some components.

In this presentation, the priority areas for nuclear data will be described. The impact of the uncertainties in nuclear data and neutron diagnostics on the design for reactors will be presented in detail.

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Submitted by **LOUGHLIN, Michael** on **Friday, December 15, 2023**

Abstract ID : **7**

Investigations into JET Soft Waste generation, speciation and detritiation

Content

Investigations into the generation, detritiation and speciation of soft waste were undertaken at UKAEA in collaboration with ITER. Soft waste arisings since 1992 were analysed to understand the mass, volume, number and tritium activities of waste packages, as well as any optimisations that could be made to operations to reduce waste generation. Experimental trials were held on soft waste items generated post DTE-2 in both air and argon to determine the effectiveness of non-destructive heating as a detritiation technique. The findings for air were varied, whereas argon was found to be ineffective. Finally, the speciation of the tritium within the soft waste was determined by thermally treating the waste and capturing the off-gas in a MARC-7000. It was found that most tritium in soft waste was in the form of HTO, at a ratio of 12:1 with HT.

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Status: SUBMITTED

Submitted by **STOKES, Thomas** on **Friday, December 15, 2023**

Abstract ID : **8**

Overview of F4E activities

Content

An overview of Fusion for Energy's (EU-DA) activities in fusion neutronics related to ITER is presented. These activities include the development of computational methods and tools such as F4Enix and F4E-Radwaste, the generation of simulation models for ITER, and analyses on design and performance issues within the ITER collaboration framework. Additionally, complementary efforts relevant to ITER are undertaken, such as radiation transport assessments for JT-60SA, nuclear data development and validation and verification (V&V) using the JADE tool within the JEFF and FENDL nuclear data frameworks. Finally, possibilities for collaborations, traineeships, upcoming workshops, and contracts are also outlined.

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Comments:

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Submitted by **LAGHI, Davide** on **Wednesday, December 20, 2023**

Abstract ID : **9**

Investigation of OpenMC for nuclear analysis of ITER

Content

ITER is a major international endeavour on the pathway to commercial magnetic confinement fusion. In 2012, ITER was formally decreed a nuclear installation (INB 174), regulated by the French nuclear safety authority, the ASN. Demonstration of the overall safety of the device is fundamental to the design, construction, operational and decommissioning phases. Nuclear analysis, or neutronics, provides a critical transverse function across the lifetime of ITER, providing input through accurate mapping of the complex pervasive radiation environment. The nuclear analysis conducted to date has been performed with some of the most high fidelity radiation transport models likely to have ever been produced. The size of the models has necessitated users to develop modifications to the source code of the reference transport code, MCNP, to reduce the computational demand and allow simulation of nuclear responses on practical timescales. Even with these improvements, ITER calculations can take of the order of weeks over thousands of CPU-cores for the most complex deep shielding problems. In this work, we present an overview of an initial investigation into the application of the OpenMC Monte-Carlo code to the nuclear analysis of ITER. OpenMC is an open source code with high scalability, support for neutron-photon coupled transport with both CSG and CAD based tracking, and is backed by a dynamic development community. We present findings on the performance, advanced code features necessary for application to this field, and calculate typical ITER nuclear responses. A comparison is made to MCNP with view to outlining outstanding gaps and development opportunities for application of this code to the nuclear analysis of ITER and other fusion devices with high level of design maturity.

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Submitted by **VALENTINE, Alex** on **Wednesday, December 20, 2023**

Abstract ID : **10**

Advanced Breeding Blankets neutronic designs and assessments for DEMO and HELIAS

Content

Advanced Breeding Blankets (BB) designs are being developed among the FP9 Horizon Europe Eurofusion Programme for the DEMO tokamak as well as for an Helical-Axis Advanced Stellarator (HELIAS) power plant.
In particular, novel designs based on modified Helium-Cooled Pebble Bed (HCPB) BB versions named (1) Water Lead Ceramic Breeder (WLCB) BB and (2) cantilever HCPB BB without First Wall (HCPB no-FW) variants have been developed under the Work Packages DEMO Design (WPDES) and Breeding Blankets (WPBB). Both concepts have been analyzed and improved under the neutronic perspective (tritium breeding and shielding functions).
Furthermore, among the framework of the Prospective R&D Breeding Blankets (WPPRD-BB) programme significant progress has been achieved in the development of a Dual Coolant Lithium-Lead (DCLL) BB for DEMO toward an advanced high-temperature (HT) concept with Single-Module-Segment (SMS) architecture (3). Such concept employs as novelty a ceramic structure in contact with the PbLi breeder and fully decoupled from the Eurofer external box by an inert gas gap. The ceramic channels allow the PbLi to go over the 550 °C creep-fatigue limit imposed by Eurofer, which is instrumental to go to higher thermal efficiency (700 °C outlet T). Different ceramic materials (SiC, alumina, zircona) have been tested and analyzed under the neutronic and activation point of view.
Finally, under the umbrella of the WPPRD Stellarator Power Plant Studies (WPPRD-SPPS) programme, aims covering the engineering activities for the development of an HELIAS power plant as alternative to the tokamak mainstream, a DCLL BB for HELIAS (4) is being also developed. Such BB concept has high potentialities to answer the specific challenges posed by the complex 3D twisted HELIAS configuration, being the breeder in liquid form and having decoupled first wall and breeder cooling circuits. Neutronic tools, analyses, integration activities and advanced solutions have been proposed and addressed as the use of a decoupled FW based on Capillary Porous System (CPS).
A comprehensive overview of the four advanced BB designs and neutronic analyses will be presented.

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Submitted by **PALERMO, IOLE** on **Thursday, December 21, 2023**

Abstract ID : **12**

Overview of STEP nuclear analysis

Content

STEP (Spherical Tokamak for Energy Production) is an ambitious UK programme that aims to demonstrate the ability to generate net electricity from fusion, targeting the completion of a prototype power plant in around 2040. The programme is currently in the concept design phase and plans to move forward to the detailed design of a single concept from 2024 onwards.

During the concept design phase of the project, it was necessary to analyse many plant concepts in a fast and efficient way. This led to the development of several tools, including paramak which allows for the generation of a tokamak CAD from script, and the use of codes such as DAGMC and OpenMC which lend themselves to automated workflows. These allowed for global nuclear quantities, such as TBR and nuclear heating, to be estimated quickly and feed into decisions on which concepts to take forward. As the STEP programme has progressed through the concept design phase, the concept stabilised and the focus of the nuclear analysis shifted, allowing the creation of localised high-fidelity models for areas of the plant that have key neutronic performance requirements.

This talk will give an overview of the STEP programme and the current design point. It will also present the nuclear analysis workflows developed for concept analysis and how these have been updated to incorporate a greater level of modelling detail within the analysis.

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Submitted by **EADE, Tim** on **Wednesday, December 27, 2023**

Abstract ID : **13**

Fusion Neutronics Challenges for ITER Analyses

Content

As the first of its kind in size and complexity, ITER represents challenges in almost every phase of design and to nearly every industry with which it interfaces. From pushing the boundaries of manufacturing to challenging the limitations of modern modeling software, ITER is creating as many questions as answers. With so many competing needs and limitations, it comes as no surprise that issues often arise that necessitate pushing yet another boundary to develop the tools needed to make ITER successful. One such area is in radiation transport. Historically, MCNP models representing a fraction of the machine, like the C-Model, have been used to assess the radiation condition in several ports around the machine. In recent years, a 360-degree model has been developed both for the tokamak as well as the surrounding structure of the building. The development of these models, specifically of the E-lite model, has revealed the true impact of crosstalk between ports—especially those in proximity to the neutral beam injectors (NBIs). The identification of this crosstalk, which is the influence that one port exerts on the radiation condition of surrounding ports, has fueled discussions about the responsibility of port integrators and of what can be done to limit the influence of nearby ports on each tenant within the ports. It is important that models which effectively capture crosstalk are accessible and easy to use while still preserving important features of the environment.

In addition to the impact of crosstalk between ports, another aspect pertinent to design and analysis of ITER is the commitment to specific design parameters. Quantifications such as shut-down dose rate (SDDR) and occupational radiation exposure (ORE) are dependent on the irradiation scenario employed by ITER. Changes to the neutron fluence could have immense impact on these parameters and affect where and how dose reduction methods are employed.

In this presentation, some of the challenges for ITER neutronics analyses are highlighted with respect to several recent ITER analyses performed by the Oak Ridge National Laboratory (ORNL) Radiation Transport team.

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Submitted by **GODSEY, Kara** on **Thursday, December 28, 2023**

Abstract ID : **14**

Key neutronics outcomes of DT campaigns at JET for ITER nuclear operations

Content

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October 2023).

An outstanding amount of nuclear fusion-related data has been collected through EUROfusion's technological exploitation of JET Deuterium-Tritium (DT) operations, which began in the JET3 project and is presently ongoing within the 'Preparation of ITER Operations' (PrIO) work-package. Several technology-driven experiments and analyses were accurately planned and executed. Their primary objectives were to advance the understanding of nuclear technology and safety, validate nuclear codes and data, and refine experimental techniques to optimize nuclear operations and maintenance at ITER, by taking advantage of the unique DT operations at JET. The two recent high-performance Deuterium-Tritium (DT) campaigns occurred at JET in 2021 and 2023, resulting in the production of more than $1.5{\cdot}10^{21}$ 14.1 MeV neutrons. The achieved irradiation conditions hold relevance for ITER technologies. Notably, the maximum neutron flux of ~$2{\cdot}10^{13}$ n/cm²/s at JET's plasma-facing components is comparable to that in the rear ITER blankets and is only one order of magnitude lower than that on the first wall for 500 MW pulse. Additionally, the 14 MeV neutron fluence of more than 1016 n/cm² is consistent with the expected fluence in rear ITER port

plugs at end-of-life as well as middle port plugs and rear blanket at the end of the expected ITER DT-1 phase. Several neutronics experiments were conducted to investigate the activation of real ITER materials, damage to functional materials, water activation in an operating tokamak cooling loop, and effects on electronics. Neutronics benchmark experiments, exploring neutron streaming through penetrations and shutdown dose rates, were undertaken to validate computational tools and nuclear data used in ITER nuclear analyses and to evaluate the shielding performance of flexible neutron shields. Given the re-baseline of ITER with two DT phases, the JET DT results and acquired experience hold unique and substantial importance for preparing the nuclear phase of ITER and supporting safety demonstrations for DT-2 licensing. This work provides an overview of these activities, highlighting key findings, achievements, lessons learned, identified issues, and crucial outcomes for the ITER program, licensing, and nuclear operations.

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Comments:

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Submitted by **VILLARI, Rosaria** on **Sunday, January 7, 2024**

Abstract ID : **15**

F4E-Radwaste and other open source developments

Content

F4E has developed and/or sponsored the development and refinement of computational tools to aid in nuclear assessments. These developments have become a crucial part of the nuclear workflow in F4E and in other working teams with a focus on open source to avoid duplication of efforts, enhance verification and to share knowledge across the ITER parties and the neutronics community. This contribution aims to present a subset of the recent F4E software developments. In particular, F4E-Radwaste, which obtains results like specific activity, dose rates and radwaste relevant parameters in a fine 3D mesh superimposed over the geometry of a MCNP model. The 3D nature of the results allows the organization of packages of activated materials in a way that minimizes the amount of mass classified with higher radwaste levels and is subject to more stringent regulations and expensive disposals. This contribution will also briefly present F4Enix, a set of Python tools and modules that have a wide range of applications for processing MCNP simulation files. JADE will also be presented, a tool for the automatic verification and validation of nuclear libraries with MCNP which has been recognized by IAEA as the official tool to perform V&V on the FENDL libraries.

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Presenter: CUBI, Alvaro (F4E/ATG)

Status: SUBMITTED

Submitted by **CUBI, Alvaro** on **Monday, January 8, 2024**

Abstract ID : **16**

Calculation of Magnet Heating Profiles Using Advanced Unstructured Mesh Variation Reduction Techniques

Content

At Commonwealth Fusion Systems (CFS), we are developing the SPARC fusion device, a high-field, compact tokamak designed to achieve net energy gain to demonstrate commercial viability of fusion energy [1]. SPARC employs advanced high-temperature superconducting (HTS) magnets that enable stronger magnetic fields to maintain the plasma. These HTS magnets must be cooled to low cryogenic temperatures to function as designed.
Specifically, the Toroidal Field (TF), Central Solenoid (CS), and Poloidal Field (PF) coils will receive a significant amount of energy deposited in the coils during a full power deuterium-tritium (DT) plasma pulse. These coils must be cryogenically cooled to prevent magnet quenching.
We are innovating in the fusion neutronics space to support detailed energy deposition calculations to inform the design of the cryogenic magnet cooling systems. We are using Silver Fir Software (SFSW)-developed tools to create unstructured meshes [2] for Monte Carlo N-Particle (MCNP) [3] radiation transport simulations. Our process involves translation of Computer Aided Design (CAD) models developed by design engineers into unstructured mesh files for geometric representation in MCNP, using SFSW meshing and automation tools. These tools include a part-by-part mesher, which allows for modularity in generating an unstructured mesh, as well as a Python Application Programming Interface (API) for full MCNP model generation.
This approach is coupled with variance reduction techniques to support modeling fine, detailed mesh of the TF, CS, and PF coils in less compute time. Specifically, SFSW-developed tool "Cottonwood" is used to generate weight-windows for unstructured mesh [4]. This allows us to leverage our existing unstructured mesh workflows that are used to support the design of the SPARC device.
In this work we take a simple energy deposition calculation and apply advanced tools to provide meaningful data to the CFS design teams.

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Submitted by **JOHNSON, Amanda** on **Monday, January 8, 2024**

Abstract ID : **17**

OpenMC Fusion Benchmarks: Streamlining Fusion Neutronics Validation and Collaboration

Content

This research introduces OpenMC Fusion Benchmarks, an automatic repository for validating the OpenMC neutronics simulation code in fusion applications. Leveraging diverse fusion benchmarks, the repository combines computational simulations with experimental data, facilitating a robust validation process.
The repository provides user-friendly functionality, enabling users to seamlessly install the package, choose a benchmark model, and execute simulations with a single command line. The results are automatically stored conveniently in a designated database folder. Utilizing Jupyter notebooks, users have the capability to visualize and compare all available results in the database.
The primary advantage of this automated approach lies in its ability to continually test and validate. Users can search for validation results for a specific combination of OpenMC version and nuclear data library, and they can actively contribute their simulation results. This approach fosters a more efficient and community-driven validation process, decentralizing the computational power required for testing new code versions or nuclear data libraries. The examination of newly submitted results allows reviewers to identify errors and bugs in new code versions effectively.
The project aims to identify areas for improvement, validate OpenMC's predictive capabilities, and contribute to the ongoing development of neutronics software for fusion. The outcomes include a valuable resource for the fusion community, enhancing the accuracy of neutronics simulations for future fusion energy systems. Additionally, the repository features an evolving database of results, ensuring continuous updates and fostering collaborative efforts in the advancement of fusion neutronics simulations.

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Comments:

Submitted for Stefano as he was having some technical problems with online submission —-Subhash

Status: SUBMITTED

Submitted by **PUTHANVEETIL, SUBHASH** on **Tuesday, January 9, 2024**

Abstract ID : **18**

Applications of F4Enix to scoping studies on W FW, borated water and blanket SB

Content

Within the ITER rebaselining exercise, the replacement of beryllium (Be) First Wall (FW) by tungsten (W) FW is foreseen. Moreover, at the March 2023 Technical Coordination Meeting (TCM) the investigation of borated water in Vacuum Vessel (VV) – Primary Heat Transfer Systems (VV-PHTS) has been requested. Last, in the perspective of using High Temperature Superconductors in the magnets of a future machine, higher magnets temperatures could be tolerated and thinner blanket modules could be used for shielding. In this context, a series of scoping studies was made by using ITER C-Model, to assess the impact of these design changes on nuclear heating and energy spectra among others. The need to perform a large number of simulations and their post-processing within the framework of these scoping studies was the perfect occasion to use and test F4Enix, a Python package for Monte Carlo simulations input and output files parsing developed at F4E. This presentation will show and discuss the results of the scoping studies and will show how F4Enix was used in pre- and post-processing to optimize the general workflow.

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Presenter: BITTESNICH, Alberto (Fusion for Energy)

Status: SUBMITTED

Submitted by **BITTESNICH, Alberto** on **Wednesday, January 10, 2024**

Abstract ID : **19**

Conversion and Performance Optimization of the ITER E-lite Model with OpenMC

Content

Over the last five years, the OpenMC Monte Carlo code has found increasing use in the fusion neutronics community thanks to continual feature additions such as coupled neutron–photon transport, variance reduction, support for CAD-based geometries, and activation/decay gamma sources that support shutdown dose rate workflows. Recently, we have begun exploring the feasibility of using OpenMC in support of ITER neutronics calculations. The MCNP model of ITER E-lite was converted to OpenMC's native representation using a Python script that handles automated conversion of MCNP models. This script has since been properly packaged into the openmc_mcnp_adapter tool. While the converter doesn't handle source definitions, tallies, and some geometry features, it does handle a sufficient subset of MCNP's syntax to successfully convert the E-lite model. Once the materials, surfaces, and cells/universes were converted, a source definition was manually constructed to match the SDEF card from the MCNP model. When simulated with OpenMC, the observed lost particle rate exactly matches that reported for the MCNP model, which underscores the robustness of the geometry converter.

With the E-lite model successfully converted, initial simulations were carried out to profile the execution speed of OpenMC. Profiling revealed that nearly all the execution time was spent in evaluating point containment (given a point in space, determine whether it is contained in a specific cell). This led to a restructuring of the algorithm in OpenMC for evaluating point containment, ultimately leading to a 4.5x improvement in execution speed. In addition to improving speed during particle transport, the point containment optimization has resulted in much better performance for visualization tasks. In particular, the openmc-plotter GUI application that is commonly used for interactive exploration of a model relies on rasterization, which in turn relies on point containment evaluations at the spatial locations corresponding to each pixel. For rasterization, the point containment optimization resulted in a 6x reduction in the time to render a 2D raster plot.

Our initial work with the OpenMC E-lite model has focused primarily on performance profiling/optimization and interactive visualization. Continued improvements in the visualization infrastructure in OpenMC can be of significant benefit to nuclear analysts; at present, the OpenMC E-lite model already has a model loading time of under 1 min, substantially reduced memory usage relative to MCNP, and with the OpenMC plotter application 2D raster plots can be generated fairly quickly. Complementary work is currently underway to compare physical tally results between the MCNP E-lite model and the converted OpenMC version.

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Status: SUBMITTED

Submitted by **ROMANO, Paul** on **Friday, January 12, 2024**

Abstract ID : **20**

Boron carbide ceramics as an in-vessel shielding material: the long road from concept to mass production

Content

In 2013, the BINP, together with RFDA and IO, started the integration of the diagnostic ports (EP11, UP02, UP08 1) and quickly came to the conclusion that without additional neutron shielding, the radiation protection requirements could not be met. It was proposed to use boron carbide for neutron shielding in the ITER vacuum chamber. This material has a low density and high neutron interaction cross-section. The material was originally proposed to be used as a powder in boxes. This concept was rejected by the IO vacuum section. This was followed by a study of methods of creating boron carbide ceramics, their properties and methods of fixing the ceramics in diagnostic ports. Numerous tests have been performed on the chemical, thermal and vacuum properties of various ceramics [1-5]. Compliance with the ITER common requirements for impurities and outgassing rate was demonstrated.
The BINP, RFDA, and IO cooperatively prepared and agreed on a specification for boron carbide for diagnostic ports (ITER_D_457TBH). There are very strict requirements for ceramics: the outgassing rate is limited to 1·10-8 Pa·m3·s-1·m-2 for hydrogen at 100 ℃.
Also at the Budker Institute, the ceramics were subjected to prolonged neutron exposure to study the degradation of mechanical properties and demonstrate resistance to cracking. The experiments were conducted on a proton accelerator developed for boron-neutron-capture cancer therapy [6].
For the Equatorial port #11, the Upper ports #02, 07 and 08, which are being integrated by the Budker Institute 1, sintered ceramics produced by Virial Ltd. were selected. Supplier were agreed upon and the serial production of ceramics was launched, including the processes of ceramics cleaning by the manufacturer according to the requirements of ITER.
Vacuum tests of supply batches of ceramics were conducted, which showed compliance with ITER Vacuum Handbook [7] and ITER_D_457TBH specification.

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Submitted by **SHOSHIN, Andrey** on **Wednesday, January 17, 2024**

Abstract ID : **21**

JMCT and its application in CFETR

Content

Joint Mont Carlo neutron photon-electron Transportation code (JMCT1) is self developed by Institute of Applied Physics and Computational Mathematics(IAPCM). It has two ways of geometry modeling, one is a 3D visual pre-processor JLAMT and the other is a CAD to Monte Carlo Geometry Converter tool (CMGC[2]). JMCT has various variance reduction techniques, powerful parallel computing capability. It has been widely used in Chinese fission reactor projects since 2013. Under support of national magnetic confinement fusion energy research project from 2015, JMCT is used to crosscheck blanket neutronics of China Fusion Engineering Test Reactor (CFETR). At the first stage, a 200MW CFETR neutronics benchmark[3] in MCNP input format based on detailed CAD design are shared in China. It is then converted to JMCT input format by use of JLAMT. The blanket modules of the benchmark are then replaced by 3 kind of blanket design concepts such as helium cooled ceramics , water-cooled ceramics , helium cooled lithium lead respectively. Another two fusion fission hybrid blankets are also investigated on the base of the benchmark. All the JMCT cross check results are in good agreement with that of MCNP and will be reported in the 15th ITER neutronics meeting. At the second stage, a 1000MW CFETR benchmark based on up to date CAD files in STP format are converted to JMCT input file by CMGC directly and the validation of the calculation is underway.

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Submitted by **SHI, Xue Ming** on **Thursday, January 18, 2024**

Abstract ID : **22**

Radiation conditions improvement in ITER tokamak complex due to leakage through penetrations

Content

ITER is a nuclear facility with 500MW of DT fusion power giving a very high flux on the first wall of the order of 1014n/cm2/s. This large specific source gives rise to several nuclear issues related to unusual radiation conditions for equipment (radiation compatibility) and potential occupational radiation exposure to workers. To ensure the targeted nuclear performance of ITER, several large openings for sub-systems are required in the bio-shield and beyond in the shielding structures. Radiation leakage through these sub-system openings can create unacceptable levels of radiation inside the tokamak complex buildings for equipment and radiological zonings defined for human safety. In order to comply with the radiation limit, a proper shielding need to be installed.
The objective of this task is to support detailed design (CDR, PDR, MRR) of several kinds of openings, performing nuclear analysis justifying opening design and proposing shielding design/materials with the objective to respect the shielding requirements based on ALARA approach (cost optimization).
Several nuclear analyses has been performed to justify the shielding for all modes: mode 0, 1, 2 and will be reported in the meeting. These nuclear analyses pertains to port-cell lintels, heavy nuclear doors, penetrations in the external walls of building B11.

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Status: SUBMITTED

Submitted by **AGARWAL, Jyoti** on **Thursday, January 18, 2024**

Abstract ID : **23**

ITER Hot Cell – A complete radiation environment assessment to support its design and operation

Content

ITER Hot Cell will probably be the largest and most complex facility of its kind. Its size, the variety of radioactive components to be managed, the sophistication of its remote-handling equipment, and the high number and complexity of maintenance operations under radiation conditions to be performed, make ITER Hot Cell's design and operation major technical challenges that directly impact the Project. How well the Hot Cell is designed will dictate how well the machine can be operated. Hence, its design cannot be decoupled from the Project's operation program, schedule and objectives.

Although its design has been addressed in the past, the Radiological Zoning and the Occupational Radiation Exposure for workers protection has only been partially evaluated. Recent changes in the Project have impacted the Hot Cell requirements and, hence, its layout design and maintenance program. Still, the radiation assessment remains incomplete to guide further steps.

In this context, we present a complete nuclear analysis of ITER Hot Cell during all operational phases: tokamak operation, transportation of In-Vessel components, and maintenance. Special emphasis is done on the latter as it is the primary contribution to the Hot Cell radiation environment. Considering the latest baseline designs from 2021, the nuclear analysis presented aims to guide further steps in the Hot Cell design. In this study we discuss in detail the following points: (i) Radiation maps of the maximum radiation sources inventory for Radiological Zoning justification. (ii) The first evaluation of the Occupational Radiation Exposure associated to Hot Cell maintenance. (iii) The layout optimization to meet radiological requirements.

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Status: SUBMITTED

Submitted by **MARTÍNEZ ALBERTOS, pablo** on **Monday, January 22, 2024**

Abstract ID : **24**

FENDL: Current status and plans for the future

Content

The Fusion Evaluated Nuclear Data Library (FENDL), whose development is coordinated by the IAEA, provides qualified nuclear data to support fusion research in general and the ITER project in particular. Recently, a new version of the library, FENDL-3.2b, has been released that underwent comprehensive validation performed by the FENDL collaboration.
This contribution will give a summary of the recent developments within the FENDL project that found their way into the latest version as well as planned future activities to improve the library. These activities also include the development of computational tools to make the library update process more robust and to automate/streamline the validation and performance assessment of the library.

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Presenter: SCHNABEL, Georg (IAEA)

Comments:

on behalf of the FENDL collaboration

Status: SUBMITTED

Submitted by **SCHNABEL, Georg** on **Monday, January 22, 2024**

Abstract ID : **25**

Recommendations for nuclear heating calculations in support of ITER

Content

Following the established workflow of nuclear analyses in support of ITER, the nuclear heating results are normally obtained in a structured rectangular mesh covering the complete geometry of interest, which serves as the input data for the subsequent engineering analyses, usually carried out using an unstructured mesh conforming to the geometry studied. In order to explore the unstructured mesh capability of MCNP6 and MCNP6-based D1SUNED code, an example of a fairly simple geometry of ITER port-mounted bolometer has been used to carry out an in-depth analysis comparing three different approaches to calculate the nuclear heating. The results obtained with the unstructured mesh have been compared to the nuclear heating of the CSG geometry obtained with Cell-Under-Voxel approach in the structured rectangular mesh and the nuclear heating calculated for a specific material in the structured rectangular mesh. This presentation provides an overview of these three different methodologies for nuclear heating calculation, discusses advantages and disadvantages of each one and gives recommendations on which approach is the most suitable in each use case.

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Comments:

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Submitted by **KOLSEK, Aljaz** on **Wednesday, January 24, 2024**

Abstract ID : **26**

Vacuum Induction Melting as a detritiation technique

Content

As part of Implementation Agreement 25 between UKAEA and ITER Organisation the Fuel Cycle Division at UKAEA has been investigating the potential of Vacuum Induction Melting (VIM) as a detritiation technique. The work started with a review into different melting technologies indicating VIM was the preferred solution and a semi-industrial scale demonstration melter was procured and installed at the FTF Rotherham site to enable trials to begin. Planned materials for testing include tritiated metals from JET (stainless steel, Inconel, Copper etc) under vacuum conditions and with partial backfill of various gases. Melt trials with a number of fusion relevant metals have been undertaken to assess secondary waste production, energy usage, size reduction requirements and operability of the furnace and results of these will be presented here.

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Status: SUBMITTED

Submitted by **REYNOLDS, Stephen** on **Wednesday, January 24, 2024**

Abstract ID : **27**

Neutronics studies on parametrized magnetic confinement fusion reactors using Serpent2 Monte Carlo code

Content

Detailed attention must be given to neutronics issues, as diverse reactor candidates are continually evaluated for their potential as future fusion power plants. Newer designs, such as spherical tokamaks, are under construction, while developments in stellarators and inertial confinement fusion (ICF) concepts progress in parallel. Over the years various CAD-based neutronics tools have been developed to facilitate the study of realistic reactor models using Monte Carlo codes. Recently, parametric tools like Paramak and HeliasGeom have emerged allowing rapid generation of reactor geometries with adjustable parameters. Despite the availability of these tools, challenges still exist in effectively simulating the most comprehensive reactor models.

This work addresses neutronics shielding and tritium breeding calculations using parametric geometry tools HeliasGeom and Paramak. The CAD models of stellarators and tokamaks, generated using these tools, are simulated using the Serpent2 Monte Carlo code. The Serpent2 provides an efficient workflow from CAD geometries to Monte Carlo simulation, enabling the direct import of geometries into the code using CAD-based STL file format. This facilitates faster design cycles as time is mostly used for running simulations and analyzing them, rather than building the geometry. Simulations of reactor models extends up to the magnetic field coils. The separation of reactor layers and coils into poloidal segments enables the production of poloidal fast neutron flux distributions, especially important for complex stellarator geometries. These distributions are used to evaluate neutron shielding capabilities of the various reactor components such as the breeding blanket.

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Submitted by **Mr LYYTINEN, Tommi** on **Wednesday, January 24, 2024**

Abstract ID : **28**

Nuclear analyses in support of the ITER Radial Neutron Camera design development

Content

Nuclear analyses in support of the ITER Radial Neutron Camera design development

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The Radial Neutron Camera (RNC) is a diagnostic system located in the ITER Equatorial Port 1 (EP01) composed by two sub-systems (i.e. In-port and Ex-port RNC 1) probing a poloidal section of the plasma through a set of fan-shaped Lines of Sight (LOS). The RNC is designed to provide a time resolved measurement of the neutron and α particles source profiles and of the total neutron source strength, through the application of reconstruction techniques to the line-integrated neutron fluxes.
The In-port RNC is embedded in a removable cassette, integrated inside the port plug Diagnostic Shielding Module (DSM) with two sets of three LOS each, probing neutrons generated in the plasma edges through single Crystal Diamond (sCD) matrix detectors and fission chambers.
The Ex-port sub-system is composed by sixteen LOS distributed in two different toroidal planes and enclosed in a massive shielding unit, extending from the EP01 closure plate through the Port Interspace, up to the Bioshield Plug. Neutrons, generated in the plasma core, stream through dedicated optical paths hollowed out in the central EP01 DSM and reach the detectors units located at the end of collimating structures. The detector unit of each LOS contains one 4He gas scintillator, one plastic scintillator as well as a sCD matrix.
Signals from the RNC detectors need preamplification because of their low amplitude. These preamplifiers, that have to be as close as possible to the detectors in order to minimize signal degradation, are hosted in a shielded cabinet located in the Port Cell (PC), that must ensure the necessary protection for the electronic devices from radiation streaming. The diagnostic system is complemented by a Service Vacuum System (SVS) providing the secondary vacuum to the In-port cassette, that is embedded in a dedicated cabinet in the PC alcove.
Specific neutronic analyses, aimed evaluating the nuclear loads to be withstood by the RNC structural elements, detectors and associated components during Normal Operating Conditions (NOC) have been performed in support of their design development. Moreover, the neutron fluxes and spectra at the detectors positions have been evaluated in order to verify the signal-to-noise ratio compliancy with the expected measurement performances of the diagnostic system.

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Submitted by **MORO, Fabio** on **Wednesday, January 24, 2024**

Abstract ID : **29**

Fusion Neutronics Analysis Capabilities in the Monte Carlo code RMC

Content

The RMC (Reactor Monte Carlo code) is a Monte Carlo transport code developed by the REAL (Reactor Engineering Analysis Laboratory) team in the Department of Engineering Physics at Tsinghua University since 2001. It has demonstrated excellent performance and has been utilized in reactor core analyses on high-performance computing platforms. To achieve such performance, RMC includes various functions like Criticality Calculation, Fixed-Source Calculation, Burnup Calculation, Shielding Calculation, various Tally functions, and more. Code originated from RMC has been effectively applied to Tokamak simulations, and RMC is fully applicable for fusion neutronics problems.

Currently, the majority of fusion neutronics models are designed in the MCNP format. To evaluate RMC's capabilities in Fusion Neutronics Problems, an input file conversion tool, namely M2R has been developed. M2R can convert inputs from MCNP format into RMC format, covering components like Geometry, Cell, Surface, and Material information. M2R has successfully converted and verified several fission reactor models, including the Hualong One (HPR1000) model. For Fusion Reactor models, M2R has also successfully converted the ITER-C Model and the CFETR Fusion Neutronics Model.

To evaluate RMC's capability in fusion neutronics analysis, we have applied it to the CFETR Fusion Neutronics Model to plot the model and calculate the neutron flux distribution. The nuclear data library FENDL3.2b was utilized in this calculation. Subsequently, we calculated the neutron flux distribution using a simply defined source. By applying Variance Reduction techniques, we determined the flux distribution on CFETR, showcasing RMC's ability for fusion neutronics analysis.

Ongoing work is mainly aimed at three planned scopes. Firstly, we will continue testing the full capability of RMC by calculating the neutron and photon flux, nuclear heating, as well as the nuclear wall loading distribution of breeding blankets in different models. We will compare these results with other MC codes such as MCNP and Serpent. Secondly, our aim is to apply RMC's CAD-based transport function to Fusion Neutronics problems, with a particular focus on the SUNIST fusion reactor model developed by Tsinghua University. Thirdly, concerning the fusion neutronics analysis function, RMC will enhance its functionalities by developing features like tritium breeding ratio (TBR) calculation and dose rate calculation. For the M2R converter tool, ongoing development efforts will focus on increasing the conversion of specific inputs, including tallies and sources definition.

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Submitted by **CHAN, KOK YUE** on **Thursday, January 25, 2024**

Abstract ID : **30**

Progress on neutronics design and analysis in SWIP

Content

The neutronics analysis and design is indispensable and strongly impacts the key performances and safety features for fusion reactors. The main progress of neutronics design and code development in SWIP (Southwestern Institute of Physics) were displayed respectively. Based on the ITER (International Thermonuclear Experimental Reactor) project, neutronics modeling and analysis (including neutron/gamma fluxes and SDDR) for China HCCB (Helium-Cooled Ceramic Breeder) TBS (Test Blanket System) in PD (preliminary design) phase was performed, including flux (neutron and photon), TPR (Tritium Production Rate), nuclear heating, SDDR (Shut-Down Dose Rate) and activation of structural materials. Based on the latest profile and layout configuration of CFETR (China Fusion Engineering Test Reactor) TBB (Tritium Breeding Blanket), the progress on the neutronics and shielding design of the HCCB TBB developed by SWIP was presented. After several rounds of neutronics design and optimization, the TBR of HCCB TBB is 1.177, and the value drops to 1.101 considering the influence of auxiliary systems, which can meet the design objectives of CFETR. The code development covers modeling and visualization, transport and SDDR calculation, optimization and so on. Firstly, to meet the demand of rapid neutronics modeling, a parametric modeling code for fusion reactors was developed, which can quickly generate neutronics models of blankets, plasma, vacuum vessel and other components. Then, the global tritium breeding characteristics and efficient optimization methods of were studied, and the multi-physics coupling and neutronics intelligent optimization code MCINO was developed. Based on this, considering the functions of tritium breeding, heat removal, shielding, structural integrity, engineering feasibility and compatibility, the intelligent and efficient optimization design of global TBR of fusion reactors could be realized. Then, the code for fusion reactor SDDR analysis based on the D1S (Direct-one-step) method with the function of mesh tally was developed and verified. Besides, a hybrid Monte-Carlo-Deterministic particle-transport code for the simulation of deep-penetration problems has been developed and preliminarily applied in the fusion field in cooperation with relevant universities in China.

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Status: SUBMITTED

Submitted by **QU, Shen** on **Thursday, January 25, 2024**

Abstract ID : **31**

Overview of IN DA Neutronics Activities

Content

The neutronics group at the ITER-India is carrying out several fusion neutronics activities for ITER Diagnostics systems (Equatorial Port Plug EPP-11 and Upper Port Plug UPP-09) as part of India's in-kind contribution. The activities include development works on comprehensive nuclear and shielding analyses supporting significant aspects of the design, optimization, engineering, and safety of systems and facilities.

Neutronics analysis for the X-Ray Crystal Spectrometer (XRCS-survey) diagnostics system at EPP-11 of ITER is being conducted at ITER-India. The preliminary shielding design for the spectrometer with Shielded Cabinet (SC) is completed. Due to space as well as weight constraint robust optimization for shielding design is carried out. Different sets of calculations have been done using MCNP. Reduced Neutron flux at the SC is achieved by optimizing the geometry of the neutron beam dump. The results are validated by the Port Integrator and IO.

The neutronic analysis for the UPP-09 was carried out for the PDR activities. The total neutron flux was calculated with an error of less than 10% for all mesh voxels. The estimation of the neutron and photon flux and nuclear heating were made. The nuclear activation analysis was done by using 175 energy groups and the FISPACT code for the estimation of the activity levels in different DSMs. The design is being further optimized for the effectiveness of the shielding using DSM by analysing the modified design and shielding in the UPP enclosure.

Nuclear activation analysis along with Radwaste classification and important radiological responses for all the components are performed for XRCS Survey diagnostics in EPP-11. The details of the neutronic analysis and results shall be presented during the meeting.

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Submitted by **Ms GAJJAR, BHOOMI** on **Thursday, January 25, 2024**

Abstract ID : **32**

Nuclear Data Uncertainty Propagation to Neutron Diagnostics

Content

The quantification of uncertainties in particle transport simulations has an important role in estimating the uncertainty in calibration, and therefore, the uncertainty of measurements performed with neutron diagnostics.
In order to quantify the effect of the uncertainty in nuclear data on neutron diagnostics, a simplified model of the ITER tokamak vacuum vessel was constructed. This model was developed using a python tool for constructing toroidally symmetric CSG geometry. Nuclear data samples were created with SANDY for isotopes Fe-56, Cr-52 and O-16. OpenMC was used to simulate particle transport in 5 scenarios focused on the neutron activation system (NAS) upper-port (UP) 18, since this diagnostic is crucial for in-vessel neutron calibration. These 5 scenarios consist of DD and DT volumetric plasma sources and a point isotropic source at 3 positions, representing a toroidal scan with a neutron generator (NG).
It was calculated that the uncertainty in reaction rate in In-115 for NAS UP18 is 0.3 %, 0.2 % and 0.1 % due to an uncertainty in nuclear data for Fe-56, Cr-52 and O-16, respectively. When performing a toroidal scan at R=598.5 cm and Z=120.1 cm, the uncertainty due to nuclear data is approximately the same when the NG is in line with the equatorial port (EQ) 18. When the NG is rotated by 60 degrees toroidally the relative uncertainties due to nuclear data increase by a factor 2-3, depending on the isotope and side of the diagnostic first wall (DFW). This can be explained by a larger effective material thickness when the source is not in line with the DFW.
The simplified model which was developed can be used to propagate uncertainties to other neutron diagnostics, with minor modifications. Due to its parametric description, it can also be used to study the effect of geometric properties on neutronic measurements.

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Status: SUBMITTED

Submitted by **FORTUNA, Mark** on **Thursday, January 25, 2024**

Abstract ID : **33**

Experimental observation and integrated modelling of proton-beryllium fusion in He and D plasmas at JET

Content

Prior to recent changes in ITER's research plan, it was envisioned that plasma operation in ITER's pre-fusion power operation phase will be based on hydrogen and helium plasmas to study plasma start-up, commission the machine and diagnostics, and perform early physics studies, such as L-H transition experiments. These plasmas are in principle not expected to produce radiation, e.g. neutrons and gammas. However, because of the planned plasma heating scenarios, e.g. synergistic NBI and RF or 3-ion RF schemes, energetic particles in the MeV energy range are expected to be produced. At such energies minority ions, such as H or ^{3}He, can trigger fusion with beryllium impurities, which would be intrinsically present if a metallic Be/W wall would be employed at ITER. In this contribution we will present the experimental and modelling effort to study these reactions in ITER-relevant plasma conditions at JET, and develop and validate a methodology for the calculation of realistic neutron sources. These sources will enable further analyses of expected levels of neutron and gamma radiation and propagation of these to compute irradiation maps, and synthetic diagnostics which can be used for diagnostics calibration. In the presentation we will elaborate on the following main achievements of the project:

• Re-evaluation of relevant proton-beryllium fusion cross sections, which were sent to IAEA for inclusion in the ENDF library.
• Execution of dedicated proton-beryllium JET experiments in ITER PFPO-relevant plasma conditions (He and D campaigns), in which a sustained proton-beryllium fusion yield was achieved unambiguously for the first time, thoroughly supported by a suite of fusion product diagnostics.
• Development and demonstration of a proton-beryllium fusion product modelling workflow using TRANSP, LOCUST and DRESS, which includes charged and neutral fusion products for the two-stage proton-beryllium fusion reaction chain.
• Integration of proton-beryllium reactions in DRESS and calculation of fusion product characteristics, providing input for charged fusion product modelling and assessing the neutron emissivity profiles and energy spectra for individual proton-beryllium reactions.
• Compilation of an MCNP SDEF proton-beryllium neutron source and demonstration of neutron transport in JET's MCNP model, computing the neutron fluxes and neutron energy spectra for selected neutron detectors.

We further showcase a solid match between the measured and computed total neutron rate in the JET proton-beryllium experiments, validating the first of a kind integrated modelling approach and demonstrating the ability to realistically model a variety of ITER-relevant fusion sources, including those expected in a full-tungsten wall machine.

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. This work has been part-funded by the EPSRC Energy Programme with grant number EP/W006839/1. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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Comments:

As communicated to the organizers I would kindly ask for the possibility to give my presentation remotely, since my family situation does not enable me to travel at this point. Thank you for your understanding.

Status: SUBMITTED

Submitted by **Dr ŠTANCAR, Žiga** on **Thursday, January 25, 2024**

Abstract ID : **34**

Development of fusion neutronics tools at ASIPP and its application on CFETR radwaste assessment

Content

The materials in the fusion system will be activated by the neutrons generated from D-T fusion reactions. The accurate estimation of the quantity and the radioactivity level of the radwaste plays an important role in radwaste minimization. A workflow of a series of software and toolkits is developed and used in the Institute of Plasma Physics, Chinese Academy of Sciences (ASIPP) to perform radwaste analysis for fusion facilities.

The quantity and radioactivity level of radwaste produced in the Chinese Fusion Engineering Testing Reactor (CFETR) is estimated with the workflow. A detailed 3D model for CFETR is created with cosVMPT from the CAD model with supplement design information. The NATF, a code coupling Monte Carlo transport code and FISPACT-II is developed to perform data analysis, pre- and post-processing, and classification of the radwaste under supported classification regulation. The part-cell-list that defines the relationship of the component/part to cells can be obtained by a scanning process by NATF. An optional cell split can be performed with NATF to cut large cells into smaller ones with specified resolution, to get more accurate neutron spectrum distribution. The part-cell-list will automatically update when cells are split. The cell list of all the non-void cells, or just the cells of interested components is obtained by a scan process by NATF, allowing users to calculate their volume by stochastic estimation with ray tracing. The NATF then extracts the volumes of wanted cells and writes the tally card for neutron flux calculation. The neutron flux of the components of CFETR is then calculated with Monte Carlo code with OTF-GVR accelerating technique. The input files for inventory calculation with FISPACT-II of each non-void cell of interested components are created by NATF with the neutron flux provided in the output of particle transport, irradiation scenario, impurity information, and part-cell-list. The activation responses, including activity, decay heat, and contact dose, are read to analysis the radwaste level with the provided regulation framework. NATF currently supports radwaste classification regulation of several countries, such as China, Russia, the UK and the US, and support for France is forthcoming.

The radwaste quantity and activity level were assessed for CFETR components. Several methods are evaluated to minimize the quality and quantity of the activated solid waste as low as reasonably achievable via source control, recycling and reuse, clearance, optimization of the management at the designing phase of CFETR. The influence of the operation time of in-vessel components on the radwaste severity and quantity is also evaluated. The influence of impurities was estimated and the impurity control level was proposed for CFETR.

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Status: SUBMITTED

Submitted by **ZHANG, Xiaokang** on **Thursday, January 25, 2024**

Abstract ID : **35**

Applying on-the-fly (OTF) variance reduction technique to radiation transport simulations of fusion facilities

Content

Conventional variance reduction (VR) techniques applied in Monte Carlo radiation transport simulation of fusion facilities rely typically on a multi-step workflow involving user interventions, manual iterations, and optimizations. While fully functional, these approaches may not meet application requirements and may not be fully adopt standards of performance and versatility required to solve problems encountered in fusion projects. Validation and verification processes are necessary to confirm the accuracy and reliability of the developed VR techniques.
At KIT, a global VR scheme employing a flux-weighted window (WW) iteration approach has been developed for the generation of WW meshes. The on-the-fly (OTF) global VR method allows for the synchronous computation of the global flux and the WW mesh within a single run, eliminating the need for an additional code or manual iterative process. In addition, a consistent computational model is used for both WW generation and flux calculation, avoiding further approximations to the geometry or source definition.
The OTF method was verified in this work using computational models of the JET (Joint European Torus) tokamak building for deep neutron penetration problems. The simulation results of the neutron flux distributions showed good convergence with low statistical error and were compared with computational results using different methodologies and experimental data obtained during JET campaign. The OTF method has been used for neutronic simulations in other fusion facilities, such as DEMO and IFMIF-DONES (International Fusion Materials Irradiation Facility – Demo Oriented Neutron Source). The results obtained also show good agreement with previously obtained simulation results, but they were received faster, with greater accuracy, with less computational power, and the resulting WW can be used for future similar calculations.

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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Status: SUBMITTED

Submitted by **AFANASENKO, Roman** on **Thursday, January 25, 2024**

Abstract ID : **36**

Neutronic analyses of two irradiation modules for the HCPB breeding blanket inside the IFMIF-DONES Test Cell

Content

This work aims to perform neutronics analyses demonstrating the unique capabilities of the IFMIF-DONES (International Fusion Materials Irradiation Facility – DEMO Oriented NEutron Source) Test Cell (TC) for neutron and gamma irradiation testing of the functional materials of fusion breeding blankets. The radiation environment within the IFMIF-DONES TC approximates the radiation conditions predicted for the EU DEMO breeding blanket. The primary goal of IFMIF-DONES TC is to irradiate the structural materials for use in DEMO installed inside the High Flux Test Module (HFTM) up to relevant high doses. Testing of the functional materials is formulated as an additional task by utilizing the Other Irradiation Modules (OIMs). The OIMs are located behind the HFTM, in the region of the Medium Flux Test Module (MFTM) of IFMIF-DONES. The radiation exposure of OIM is attenuated by the HFTM, the total neutron flux at the front of OIM is about 8e13 n/cm2/s. At IFMIF-DONES, neutrons of a wide energy spectrum peaked at 14 MeV are produced by the Li(d,xn) deuterium-lithium nuclear reactions inside the liquid lithium target impinged by the 40 MeV deuteron beam formed by the linear accelerator. The neutron and photon radiation transport simulations have been performed using the McDeLicious-17 code package, which is the MCNP6.2 code modification to simulate the d-Li nuclear reactions.

We present neutronics analyses of two OIMs that are in the designing process: 1) Tritium Release Test Module (TRTM) 1 and 2) BLUME - Blanket functional materials module of the Helium Cooled Pebble Bed (HCPB) breeding blanket [2]. The neutronics models of the two irradiation modules have been developed and integrated into the reference TC neutronics model. Neutron and photon fluxes, nuclear heating, neutron damage (dpa), and tritium production inside the irradiation modules have been calculated. The neutronics results were normalized to the DONES-standard 125 mA deuteron beam current of the 40 MeV deuterons, with the assessment of potentially doubling the beam current. The impact of the OIM installation on the neutron spectra inside the TC-adjacent Complementary Experiments Room has been assessed as well. Previous steps of neutronics analyses are available in [3].

Acknowledgments:
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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1 K. Kondo et al., "Neutronic Analysis of the IFMIF Tritium Release Test Module Based on the EVEDA Design", Fusion Science and Technology, 66:1 (2014), 228-234, https://doi.org/10.13182/FST13-74310.13182/FST13-743
[2] G. Zhou, et al., "Solid Breeding Blankets: overview and key pending validation needs," Second IFMIF-DONES Users Workshop, Granada, Oct. 19 – 20, 2023, https://indico.ifmif-dones.es/event/4/contributions/377
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Status: SUBMITTED

Submitted by **SERIKOV, Arkady** on **Thursday, January 25, 2024**

Abstract ID : **37**

Actionable workflows for fusion neutronics simulation

Content

Neutronics simulations are complex and typically require significant effort in setting up multiple pre- and post-processing steps, such as preparing data and geometry before the neutronics simulation can be carried out. From the end user point of view, many complex tools must be mastered in order to run the simulations, creating a barrier of entry to a field that needs to transition from science to engineering. Different versions of the tools used and customisation of processing steps can also lead to reproducibility issues, and the data produced could often be better managed.
We propose setting up standard packages (i.e. OpenMC and Paramak) as interoperable tools that can be linked up to create an automated simulation process, to be executed using a workflow engine (i.e. Galaxy). The integrated tools can then be (re)configured into scalable, actionable workflows that are FAIR; findable, accessible, interoperable and reusable. The chosen workflow engine provides a simple and accessible interface with many added benefits, such as capturing metadata, documenting what simulation has been executed, when, by whom, how and why. The selected workflow engine also enables automatic scheduling on distributed and high performance computing systems.
The presentation will use a spherical tokamak case study to show how this approach can be used to orchestrate neutronics simulations. The authors will first show how individual tools can be put together as automated workflows. By presenting the results of a neutronics simulation carried out in this way, the authors will then highlight the simplicity and added benefits of workflows.
The work is aimed at the neutronics community but especially newcomers or those outside the community (such as SMEs or young researchers) who wish to run basic simulations but are unfamiliar with the tools used in the sector.

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Status: SUBMITTED

Submitted by **SMITH, William** on **Thursday, January 25, 2024**

Abstract ID : **38**

Depletion analysis and material damage on 3D ARC-class reactor

Content

In a fusion reactor, materials endure substantial fluxes of high-energy neutrons, resulting in radiation damage and triggering nuclear reactions that alter the chemical composition of materials through transmutation. Many of these reactions produce gases, notably hydrogen and helium, which further contribute to swelling and embrittlement of the materials.
This investigation employs the Monte Carlo code OpenMC with DAGMC to analyse in a 3D CAD geometry an Affordable Robust Compact (ARC) class reactor, a conceptual design for a D-T Tokamak proposed by researchers at the Massachusetts Institute of Technology. OpenMC supports depletion calculations coupled with transport and transport-independent calculations and evaluates the evolution of nuclides density inside the material. The predicted nuclide densities at a future time step are then utilized to determine updated reaction rates, and the process is repeated for all required time steps.
The D-T fusion reactions do not directly produce radioactive products, but the emission of fast neutrons can induce nuclear activation in structural materials. This study focuses on the primary aspects of neutron irradiation on solid materials in the ARC class reactor, with particular emphasis on the effect of neutron-induced activation.
Therefore, specific activity, decay heat, and shutdown dose rate are calculated. These allow for waste classification based on activity content to determine appropriate recycling and reuse strategies for discharged components and materials.
To validate the accuracy of the results, a comparison was made with corresponding results obtained using the FISPACT-II code. The agreement between the two codes serves as a benchmark for the reliability of OpenMC in predicting nuclear activation phenomena in fusion reactors.

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Presenter: PETTINARI, Davide (Politecnico di Torino)

Comments:

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Submitted by **PETTINARI, Davide** on **Thursday, January 25, 2024**

Abstract ID : **39**

Implementation of Attila4MC's contiguous mesh converting method to a DAGMC geometry for transport calculations in OpenMC

Content

The usual method for Monte Carlo transport geometry description is constructive solid geometry, which can be challenging and prone to errors for complex models such as fusion power devices. For complex geometries, the open-source transport code OpenMC supports the use of Direct Accelerated Geometry Monte Carlo (DAGMC) to represent CAD-based geometry. The traditional DAGMC workflow relies on the commercial software Coreform Cubit for mesh generation and additional steps to track particles on a CAD geometry using OpenMC. The workflow suggested in this paper takes advantage of the Attila4MC package for the creation of a contiguous unstructured tetrahedral mesh (RTT mesh). In the Attila4MC mesh generation process, the user has control over the maximum allowable edge length of a mesh and curvature refinements for a more accurate model representation. The generated RTT mesh is converted into a H5M file, which is the native file format of Mesh-Oriented datABase (MOAB) meshes. Two Python interfaces, PyMOAB and PyDAGMC, are useful for adding a few code-specific steps that are needed to add the information required to use the MOAB file as a DAGMC model in OpenMC. Lastly, the resulting H5M file from this workflow is implemented in OpenMC as both an unstructured mesh for tallies and a DAGMC geometry for high-fidelity transport calculations. This paper will discuss the development and implementation of this workflow.

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Status: SUBMITTED

Submitted by **NOVAIS, Felipe** on **Friday, January 26, 2024**

Abstract ID : **40**

Applicability analysis of FLUKA for fusion neutronics

Content

Monte Carlo methods are commonly applied in fusion neutronics for simulating particle transport calculations, due to the complex geometry and the capacity to use continuous energy cross-section data. FLUKA is a versatile particle transport calculator that specializes in radiation shielding analysis and has the potential for neutronic analysis in fusion research. Evaluation of shutdown dose rates (SDDR) holds great significance in fusion neutronics. The aim of this study is to analyze the applicability of FLUKA in the field of fusion neutronics through a comparative analysis of FLUKA and MCNP in calculating activation and SDDR. The comparison is conducted in terms of neutronics modeling, particle transport calculations, and data analysis, using the ITER shutdown dose rate experiment T-426 as the benchmark problem. In neutronics modeling, both FLUKA and MCNP use the Constructive Solid Geometry (CSG) method, but FLUKA provides an interactive modeling interface. In terms of particle transport calculation, FLUKA can obtain the neutron energy spectrum, material activity, and related dose rate in a single step, but MCNP requires assistance from other code. Additionally, FLUKA provides a plotting tool for data analysis, which offers greater convenience compared to the current mainstream multi-program coupling calculation. In terms of accuracy, this study compares the T-426 experimental data with simulation results obtained using two methods: the Rigorous Two-Step (R2S) method based on MCNP and activation coupling code, and the direct calculation in FLUKA. Comparison between the neutron energy spectrum of the target cell and the activity of each nuclide obtained from two method reveals a good agreement. The difference between the calculated and experimental SDDR using the R2S method ranged from 2.56% to 26.51%. The SDDR obtained from FLUKA also exhibit a great level of agreement with the experimental data, except for the first day after shutdown, requiring further analysis in future studies.

Keywords: FLUKA; Fusion; Neutronics; Shutdown dose; Monte Carlo

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Status: SUBMITTED

Submitted by **TONG, xilong** on **Friday, January 26, 2024**

Abstract ID : **41**

Neutronics activities for KODA diagnostics

Content

Various neutronics activities have been conducted for KODA diagnostic systems including UP18(upper port #18) integration, NAS(Neutron Activation System), and VUV(Vacuum Ultra Violet) spectrometer systems. Nuclear heat and SDDR (ShutDown Dose Rate) have been estimated for the UP18 based on the ITER C-model using D1S-UNED and SRC-UNED. The radwaste assessment activities for UP18, VUV, and NAS have been conducted, and the radiation shielding evaluation for VUV spectrometers has been conducted as well. In addition to the analyses, the radiation tolerance test of various components is ongoing, such as the neutron irradiation test of BI-CCD for the VUV spectrometers. The status of neutronic analyses and the experimental tests of various components under neutron environments will be presented.

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Status: SUBMITTED

Submitted by **AN, YoungHwa** on **Friday, January 26, 2024**

Abstract ID : **42**

Neutronics activities for A-FNS facility design

Content

The A-FNS facility, a high-current accelerator-based neutron source for fusion material development, consists of a deuteron accelerator system of CW, 125 mA, a liquid lithium target system and an irradiation test system with a few test modules to be able to perform for many fusion materials. Since the conceptual design phase in 2016 onward, the A-FNS facility design has been predominantly performed by Monte Carlo neutronic analyses. Especially various nuclear responses such as nuclear damages, heating, gases production for the fusion materials in the test system, estimation of the activation in the target system and shielding performances of the facility buildings have been carried out. We have studied for effective shielding calculations with variance reduction parameter generation code to reduce statical errors and obtain calculation results with high accuracy, and benchmarked many nuclear data libraries in order to contribute to advance the reliability and precision on nuclear data for A-FNS facility design. Here, we describe the all latest neutronics activities of A-FNS mentioned above and share a common issue on fusion neutronics community. This abstract is subject to further elaboration during the presentation. More detailed information will be provided at the workshop.

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Status: SUBMITTED

Submitted by **KWON, Saerom** on **Friday, January 26, 2024**

Abstract ID : **43**

Monte Carlo simulations for ITER Neutron Diagnostics

Content

DD and DT fusion reactions produce 2.45 MeV and 14.1 MeV neutrons respectively: the measurement of neutronic quantities, such as neutron fluence, neutron flux and neutron emission profile allows the reconstruction of the fusion power and plasma profile.

ITER tokamak will rely on a set of different neutron diagnostic systems: neutron activation system, neutron flux monitors, radial and vertical neutron cameras. The assessment of the calibration factors for each detector is one of the key points to be addressed: together with two experimental calibration campaigns, it will heavily rely on computational tools for neutron transport simulations.

The level of detail in representing complex geometry and the nuclear data uncertainties will affect the accuracy of the simulations and, consequently, the accuracy of the measured quantity.

The scope of the work is to present the overview of the needs for Neutron Diagnostics, the on-going Monte Carlo simulations and the preliminary results.

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Presenter: MARIANO, Giovanni (ITER Organization)

Status: SUBMITTED

Submitted by **MARIANO, Giovanni** on **Friday, January 26, 2024**

Abstract ID : **44**

Nuclear Analysis Requirements for Compact Fusion Pilot Plants

Content

Compact reactors constitute an important intermediate/pilot stage where, the purpose is to bridge the science and technology gaps between ITER and DEMO and hence providing a solid technical basis for power plant 1. We define pilot plant (PP) as a reactor that demonstrates an un-interrupted gross electricity production and the re-use of the tritium bred in the blanket. In this work we elucidate the nuclear analysis requirements for a compact pilot plant of about 3.6 m major radius in an aspect ratio range of 1.9 to 2.35 for a fusion power of about 300 MW and a fusion gain of about 5 [2, 3]. The average neutron wall load is about 0.75 MW/m2. The constraints posed on the nuclear analysis, arising from the operational lifetime (1 FPY) and limiting dose based on materials choice, the desired neutron shielding, HTS magnets, breeding blanket, and maintenance will be discussed. Preliminary nuclear analysis of different shielding materials and their impact on the overall reactor lifetime will be elucidated [4]. The nuclear analysis requirements for different blanket and maintenance concepts, shielding materials and their impact on the overall reactor lifetime will be presented.

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Status: SUBMITTED

Submitted by **PADIVATTATHUMANA, Maya** on **Friday, January 26, 2024**

Abstract ID : **45**

Present activities on Neutron Engineering of WCCB-TBS

Content

As a blanket neutron radiation engineering test toward the demonstration of JA-DEMO reactor power generation, QST of Japan has been planning neutronics tests using a Test Blanket System (TBS). The TBS is a type of Water Cooled Ceramic Breeder (WCCB) and QST are planning to conduct verification tests of so-called neutron engineering property such as tritium recovery and verification of nuclear heating, which are the most important aspects of fusion neutronics engineering, according to ITER operations. In this workshop, we will present the technical status of the following neutronic related items and discussed about neutronics R&D that is considered important for FPO in ITER.

1. WCCB-TBS neutronic engineering specifications,
2. Nuclear activation analysis on WCCB-TBS design,
3. Development of neutron activation system,
4. Tritium recovery experiment of fusion neutronic tests

Primary author: OCHIAI, Kenta (National Institutes for Quantum Science and Technology)

Presenter: OCHIAI, Kenta (National Institutes for Quantum Science and Technology)

Comments:

As co-authors, the following persons should be accounted, Yoshinari Kawamura from National Institutes for Quantum Science and Technology (QST) Takanori Hirose from National Institutes for Quantum Science and Technology (QST)

Status: SUBMITTED

Submitted by **OCHIAI, Kenta** on **Friday, January 26, 2024**

Abstract ID : **46**

Novashield® HE : A New generation of High Efficiency RadioProtection Materials for fusion Applications. High Hydrogen content and High fire resistance

Content

Since 1970, Lemer Pax delivers specific solutions for most demanding requirements by designing and assembling radiation protection solutions for the industrial, nuclear, research and medical sectors. We have become a provider of global solutions that bring together all the expertise along the entire value chain of a nuclear facility when it comes to radiation protection. From design to operation to decommissioning, Lemer Pax guarantees optimal operational radiation protection, total worker safety and compliance with national and international regulations. LEMER PAX exports its equipment/solutions in a significant number of countries.

ITER requested LEMER PAX to provide a solution for increasing the radiation protection around the Tokamak. ITER's Nuclear Safety division drew up the initial requirements and presented its roadmap to Lemer Pax in April 2021. In short tile, he aim was to develop highly hydrogenated borated materials, mainly to slow down and trap neutrons with boron. In other words, the main driving forces behind the request were to improve radiation protection while offering good level in fire resistance and a density as close as possible to 1.

Radiation generated by the TOKAMAK requires the installation of specific shielding in order to meet radiation protection objectives and to limit the radiative environment. It was therefore necessary to respect a strict limit as for the weight per m2 . This involved to create high fire resistance neutron-absorbing and light materials. The basic best suited compositions were identified to be :
- hydrogen-rich borated mortar family (hydrogen content ≥ 7wt%)
- and borated polymer (hydrogen content ≥ 10 wt%).

Within a stringent timeframe, Lemer Pax developed suitable formulations and associated innovative materials with enhanced performance in terms of radiation protection: the Novashield®HE (High Efficiency) family of products. These new hydrogenated materials, which are neutron-absorbing and fire-resistant, maintain their radiation protection performance against fires. They have been developed to meet all the needs of the nuclear sector. Their use is now recommended from the design phase in order to reinforce the neutron and gamma protection of the ceiling of the ITER tokamak for example. These technical solutions are designed for both fusion and fission. In particular, these recommendations led to the emergence of two new materials, the Boron enriched mortar BORATED MORTAR 075 and the borated polymer POLYBORE HE 105.

Another application was the biological screw type (Bio screw), which are not historically designed for the Neutron attenuation. In order to improve the radiation interaction surface, Airshield® Neutron was developed. It is based on a stainless steel sheath (casing) where thousands of balls made from POLYBORE HE105, which are piled up and having a specific shape.

Primary authors: Dr EL HABER, Fady; Mr LEMER, Pierre-Marie

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Presenter: Dr EL HABER, Fady

Status: SUBMITTED

Submitted by **EL HABER, Fady** on **Friday, January 26, 2024**

Abstract ID : **47**

UNED developments for the calculation of radiation source models associated with water cooling circuits in fusion installations.

Content

In water-cooled fusion installations, such as ITER (Nuclear Facility INB-174) and DEMO, the water is exposed to high-energy neutrons. This exposition leads to the generation of two types of radiation sources from the Activated Water and Activated Corrosion Products (ACPs). The evaluation of these sources and their impact on the radiological levels is necessary to demonstrate compliance with safety requirements but presents methodological challenges due to the complexity of the circuits.

In previous years, a series of neutronics studies of the ITER Tokamak Cooling Water System (TCWS) have been done using simplified or partial models and applied different methodologies. These analyses provided results with a high degree of variation suggesting a large uncertainty and posing a challenge in the safety demonstration. With this perspective, since 2020 the UNED team has developed a set of workflows, tools, and codes aimed at removing assumptions and increasing the level of detail in the neutronics analyses either for entire circuits or at the level of single components.

At the system level, a set of automated SpaceClaim scripts allowed the practical production of a detailed model, in terms of geometry and topology, of the entire TCWS outside of the Vacuum Vessel. This model was used with FLUNED-SL, a system-level code that produced the associated radiation source for the activated water by calculating the transport of the radioisotopes. A second tool was developed to rapidly apply the OSCAR-fusion ACP calculation results to the detailed TCWS model and produce the associated radiation source. Analogously to the E-lite model for neutronics, the use of a single, detailed model with a unified methodology proved beneficial for the ITER TCWS radiological analyses.

At the component level, some codes were developed to analyze in detail single components that have a major impact on the radiological conditions. For the activated water, the FLUNED code was developed to estimate the radioisotope generation, transport, and decay in components with complex fluid dynamics using a Finite Volume method. FLUNED was applied to evaluate the 16N concentration in the Vacuum Vessel and in the connected circuit, during a 500 MW ITER pulse. For the ACPs, the SURF-UNED tool to estimate the radiation field given by the ACPs deposited over an arbitrarily complex set of surfaces was developed and applied to one ITER Heat Exchanger. The development, validation, and application of these tools will help in reducing the uncertainties associated with the calculation of the radiation fields surrounding these components.

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Presenter: DE PIETRI, Marco (UNED)

Status: SUBMITTED

Submitted by **DE PIETRI, Marco** on **Friday, January 26, 2024**

Abstract ID : **48**

ITER NEUTRON SOURCE SPECIFICATION FOR OPENMC

Content

We consider the OpenMC 1 software package as a potential alternative to MCNP, which is actually the only one being used at ITER. At least, OpenMC can be a good complement for preliminary evaluations, lost particles elimination, weights calculations for variance reduction and similar tasks. OpenMC presents the necessary functionality for these tasks. Besides, it is based on modern high-performance technologies. The software project structure, style, and engineering procedures meet all modern standards. The OpenMC is available for everybody as an open-source code. This allows, if necessary, to adopt the code in the aspects, that are necessary to solve our specific tasks. The project has over 100 developers from many countries around the world, including China and Russia. NEA/OECD provides online courses on OpenMC for the growing users community.

The major issue of using OpenMC as an alternative to MCNP is the need to maintain equivalent models for both MCNP and OpenMC. As for geometry, there are several solutions for back and forth conversion of MCNP <-> OpenMC, as well as CAD <-> MCNP, CAD <-> OpenMC. These solutions still need further development and testing, but essentially the problem of supporting synchronization of model geometry is solved. The identical logic of CSG (constructive solid geometry) specification representation in both MCNP and OpenMC is a basis of feasibility of the converters development.

However, the issue with the synchronization of the neutron source specification is still open. The available implementations of the parametric neutron source of the tokamak, from our point of view, are a good basis, but do not allow us to define the neutron source adequately yet. To solve this issue, we have developed software to generate a neutron source in two versions:

- from a table with temperature and ion concentration distributions (ni(Ψ), Ti(Ψ)) as a function of the magnetic surface coordinate Ψ and parameters specifying the spatial distribution of magnetic surfaces, as presented in [2]. Like in [2], the reaction rate parameterizations from [3] are used to calculate the neutron source intensity.
- from the neutron source distribution I (R, Z), which is calculated with other tools from plasma equilibrium state.

As a result, OpenMC neutron source specifications are obtained for ITER using data from [2], for the DT scenario (400 MW), and for TRT [4]. The paper presents a comparative analysis of the results of MCNP and OpenMC calculations.

This research, its findings and results are supported by the state contract №H.4a.241.19.23.1014 as of 18th of January 2023.

ITER is the Nuclear Facility INB no. 174. The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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Status: SUBMITTED

Submitted by **AFANASENKO, Egor** on **Friday, January 26, 2024**

Abstract ID : **49**

R2S-RFDA DEVELOPMENT STATUS

Content

We continue developing software packages r2s-rfda 1, mckit [2], mckit-nuclides [3], mckit-meshes [4], xpypact [5] to provide complete activation information with reasonable performance.
The system stores input and output data in analytical database (ADB) based on DuckDB [6]. Computations include running FISPACT [7] over a large number of cases (up to several thousands). Large volume of FISPACT output requires preliminary aggregation of JSON files produced by FISPACT in parquet files. This is implemented with xpypact based on Polars [8] for large volume data processing. On the following step, the data is transformed to activation characteristics (nuclides, activity etc.) of a model cells. Besides, we compute induced gamma source spectra, and spatial distribution for given moments in time.
One of the problems is computation of a contact dose. This values computation is adequate in mode "full", which requires by orders of magnitude more computational resources than the alternative mode - "simple". We applied dose weighted averaging over cell - mesh voxel intersections masses. According to our tests, the dose results of "full" and "simple" mode as well as plain FISPACT runs now are in reasonable agreement.
We present this work to discuss its advantages and shortcomings, and future needs. For instance, we fill the need to implement interface to FISPACT using C++ API and get better data processing performance using bulk data transfer directly to the ADB.
The work is budgeted with government contract 18.01.2023 № H.4a.241.19.22.1014.

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[4] mckit-meshes, https://github.com/MC-kit/mckit-meshes
[5] xpypact, https://github.com/MC-kit/xpypact
[6] duckdb, http://duckdb.org/
[7] FISPACT, https://fispact.ukaea.uk/
[8] Polars, https://pola.rs/

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Status: SUBMITTED

Submitted by **PORTNOV, Dmitri** on **Friday, January 26, 2024**

Abstract ID : **50**

Development and validation of fully open-source R2S shutdown dose rate capabilities in OpenMC

Content

Shutdown dose rate (SDR) calculations for fusion facilities are often considered a measuring stick for nuclear analysis software stacks as they comprise a wide range of necessary analyses for the design and licensing process. Here we present the first fully open-source capabilities for SDR calculations of fusion energy facilities based on the Rigorous 2-Step (R2S) methodology. These capabilities have been implemented in the OpenMC Monte Carlo particle transport code, building on its existing capabilities while also leveraging new features that have been added to the code to support SDR calculations, such as decay photon source generation. Both cell and mesh-based workflows are available including those on unstructured meshes. Each of the individual physics components in the R2S workflow—neutron transport, activation, decay photon source generation, and photon transport—have been verified through code-to-code comparisons with MCNP6.2 and FISPACT-II 4.0. These comparisons generally demonstrate excellent agreement between codes for each of the physics components. The full cell-based and mesh-based R2S results are presented for the first experimental campaign from the Frascatti Neutron Generation (FNG) ITER dose rate benchmark problem from the Shielding INtegral Benchmark Archive and Database (SINBAD). For short cooling times, the dose calculated by OpenMC agrees with the experimental measurements within the stated experimental uncertainties. For longer cooling times, an overprediction of the shutdown dose was observed relative to experiment, which is consistent with previous studies in the literature. Altogether, these features constitute a combination of capabilities in a single, open-source codebase to provide the fusion community with a readily-accessible option for SDR calculations and a platform for rapidly analyzing the performance of fusion technology.

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Comments:

This talk should complement Paul Romano's submitted talk on the conversion of the ITER E-lite MCNP model to OpenMC. Together they provide a good overview of the capabilities, both current and planned for OpenMC development for fusion.

Status: SUBMITTED

Submitted by **PETERSON, Ethan** on **Friday, January 26, 2024**

Abstract ID : **51**

Increasing Toroidal Field Magnet Lifetime via Entrained Hydride Shielding Composites

Content

In compact tokamaks, limited space means limited shielding; hence advanced shielding compositions are of interest to minimize dose rates and maximize component lifetimes. The ENtrained Hydride Absorbing Nuclear CompositED (ENHANCED) Shield project seeks to characterize the neutronic and thermo-mechanical properties of an irradiation-stable matrix (e.g. MgO) containing highly-absorbing entrained hydrides (e.g. GdH2, HfH2). The search of a parameter space of potential compositions, however, remains computationally expensive. The present study considers a reduced-order neutronics model of a commercial tokamak concept and evaluates a wide variety of possible materials for the primary neutron shield and magnet structure, with the objective of evaluating the optimal use of ENHANCED Shield materials. Spatial variation is also introduced in the model through a concentration gradient of entrained hydrides in the primary neutron shield, and a high-fidelity REBCO magnet model. In this study, candidate materials are evaluated on the basis of nuclear heating, fluence, and gas production rates. Different weight window techniques for Monte Carlo variance reduction are also evaluated for simplified geometries representative of an advanced tokamak. The energy binning structure and performance of weight windows generated via Forward-Weighted Consistent Adjoint-Driven Importance Sampling and the Method of Automatic Generation of Importances by Calculation are determined as a function of shield composition.

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Status: SUBMITTED

Submitted by **FLETCHER, Jack** on **Friday, January 26, 2024**

Abstract ID : **52**

Assessing performance tradeoffs of DAGMC and CSG geometries for fusion neutronics models

Content

A workflow has been developed for the evaluation of the performance tradeoffs associated with Direct Accelerated Geometry Monte Carlo (DAGMC) geometry models in fusion neutron transport simulations in contrast to conventional Constructive Solid Geometry (CSG) models. The assessment encompasses a range of performance benchmarks for both geometry configurations using models of increasing complexity and varying tally constructs. The evaluated models span from circular and elliptical torus models with minimal number of surfaces to fusion reactor components and fusion reactor models like those found in Paramak 1 with thousands of surfaces. Various geometric tallies (cartesian, cylindrical, spherical, etc.) are used, encompassing local to global regions of the models. Key performance metrics, such as CPU usage, memory usage, and particle simulation speed, are compared for equivalent models in both DAGMC and CSG frameworks. The benefits and drawbacks of each approach are reported as functions of model complexity and tally construct, providing insights into the appropriate applications of DAGMC or CSG geometries for various problems. Future work aims to expand the collection of models by incorporating more complex fusion geometries and tallies, further enhancing the comprehensive assessment of these modeling approaches.

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Presenter: DU, Katelin (MIT)

Status: SUBMITTED

Submitted by **DU, Katelin** on **Friday, January 26, 2024**

Abstract ID : **53**

Status of GEOUNED code

Content

The GEOUNED code is specifically designed to convert CAD models, defined using the B-rep approach, into MC radiation transport models, defined using the CSG approach, and vice versa from MC to CAD. This code incorporates standard features commonly found in conversion tools, including decomposition, conversion, and automatic void generation. Additionally, it introduces innovative features, mainly in the automatic void generation part, which are described in this article.
GEOUNED has demonstrated successful application in highly detailed 3D models used in fusion neutronics, which are known for their complex geometries, particularly those utilized in ITER. The article includes examples showcasing GEOUNED's performance in these challenging models, as well as custom applications that highlight its flexibility in addressing non-standard problems. The code is open-source and utilizes Open CASCADE as the geometry engine, with FreeCAD serving as the Python API.

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Status: SUBMITTED

Submitted by **CATALAN, Juan-Pablo** on **Friday, January 26, 2024**

Abstract ID : **54**

Physics-Informed Neural Networks for Neutronic Heating

Content

Developing components for fusion reactors poses a complex challenge due to the need to consider a large design space and tight interdependence between different systems. Fast prediction of specific physics fields is crucial for thoroughly exploring this space, ensuring candidate designs align with the specified requirements. Traditional numerical methods fall short of meeting the demand for rapid predictions and probably offer an unnecessary level of accuracy for exploratory work. Hence, there is justification for adopting a surrogate modelling strategy to fulfill this purpose.

In this presentation, we will describe a case study that explores the use of Physics Informed Neural Networks (PINNs) to estimate the temperature field in a spherical tokamak caused by neutronics heating. The work uses OpenMC to compute the neutronic heating, one of the inputs into the PINN. The standard governing equations for heat transfer and the associated thermal boundary conditions are used for training the PINN. Validation using standard finite element simulation demonstrates alignment between the results of finite element simulations and PINNs, highlighting the predictive capability of PINNs in neutronics heating thermal analysis. The pre-trained surrogate model shows the capability of making predictions much more quickly than the finite element method.

The results demonstrate the substantial potential of PINNs to speed up simulations for engineers involved in simulation work and to accelerate the design process of components subjected to neutron interactions.

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Status: SUBMITTED

Submitted by **MIAO, ZEYUAN** on **Friday, January 26, 2024**

Abstract ID : **55**

Green's function approach for performance assessment of ITER neutron diagnostics

Content

This research serves as an extension and synthesis of findings from previously published works [1–5], focusing on modeling the process of measuring high-temperature plasma parameters in ITER using neutron flux monitors. The measurement modeling method was applied and examined within the framework of developing the Divertor Neutron Flux Monitor. Insights gained from earlier experiences were applied to two other analogous diagnostic systems for the ITER tokamak: the Neutron Flux Monitor (NFM), the Microfission chamber (MFC) and the Neutron Activation System (NAS). DNFM, NFM, and MFC diagnostics are designed to measure the neutron emission rate of fusion neutrons and the fusion power of the ITER plasma across a broad dynamic range. NAS will measure first wall neutron fluence and will provide the total number of emitted neutrons during plasma discharge.
After installation in the tokamak, all three systems will undergo calibration using standard DD and DT neutron sources. The work provides a concise overview of the considered neutron diagnostics and presents results from modeling the reaction rate of the detector's converter material in ITER experiments involving DD plasma, utilizing the Green's function of radiation fields [2].

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Presenter: KOVALEV, Andrei (ITER)

Comments:

Please, put my talk right after Giovanni Mariano

Status: SUBMITTED

Submitted by **KOVALEV, Andrei** on **Friday, January 26, 2024**

Abstract ID : **56**

Towards automatic CAD defeaturing for fusion neutronics

Content

A typical neutronics simulation process may involve the conversion of a detailed CAD model into a modified analysis model. This conversion process almost always requires prior defeaturing or modification of the detailed model's geometry, whether it is for neutronics simulation software that uses constructive solid geometry or faceted geometry.
Some simplification guidelines have been suggested. For example, conversion from a neutral CAD format (typical of design software) to constructive solid geometry requires the removal of elements such as higher-order surfaces and splines. In the case of conversion into faceted geometry, the geometry does not contain surface overlaps. A concern driving the need for automation is how this can be executed efficiently for large, complex CAD models. This is because the defeaturing process is typically done manually and usually relies on the experience and judgement of practitioners. Also, this iterative process is one that often necessitates collaboration between the designer and analyst to produce models of appropriate complexity and accuracy suitable for the task.
The presentation will outline the key challenges in automating defeaturing, describe geometric characteristics desirable for neutronics modelling, and suggest techniques that will help deliver the goal of creating an automated CAD defeaturing process. This work will be of interest to those involved with neutronics modelling who receive CAD models directly from the designer and are motivated to improve their practice. It will also benefit researchers who are looking at ways to streamline neutronics simulation procedures.

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Status: SUBMITTED

Submitted by **SOEMANTORO, Raska** on **Friday, January 26, 2024**

Abstract ID : **57**

Analysis of a Segmentation Approach to Breeder Blanket Design and the Utilisation of FLiBe as a Novel Neutron Reflector

Content

Tritium production in a fusion reactor is a requirement for a deuterium-tritium based plasma source and the design of a lithium containing breeder blanket is essential to a closed fuel cycle in an electricity to grid style power plant. Tritium is also a rare resource currently only produced in a number of fission reactors. Due to the shortfall in experimental data in this area neutronics calculation will lead the design of the first-generation fusion blankets tasked with the production of the fuel. Without such maximised breeding potential additional start up time of reactors may need to postponed due to a low global inventory of tritium. Within this study a description of an optimisation design process which segments an EU-DEMO style blanket to provide freedom in material allocation, along with the necessary validation steps required for repeatable results. The emergent blanket is used within OpenMC to test the performance of a hybrid liquid metal and molten salt breeder. Molten salts are proposed as an eligible lithium source in a fusion environment. The presentation will highlight how FLiBe can be used as a novel reflector due to its ability to breed tritium and reflect lower energy neutron back to the bulk of the breeding section. This indicates that TBR (Tritium Breeding Ratio) can be improved with differing material configurations rather than more conventional, single breeding methods providing optimum solutions that allow for more cost effective, and compact, blanket region while maintaining high TBR values.

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Status: SUBMITTED

Submitted by **Mr BARKER, Adam** on **Friday, January 26, 2024**

Abstract ID : **58**

Overview of the ongoing EUROfusion R&D activities on the ITER Activated Corrosion Products

Content

The wet surfaces of nuclear reactor systems have the potential to release Corrosion Products (CPs) into the coolant, which, under neutron irradiation, become Activated Corrosion Products (ACPs). ACPs pose a significant radiological hazard in nuclear plants, including both fission and fusion facilities. The activated products, transported by the working fluid, reach ex-vessel regions accessible to personnel during maintenance, thus emphasizing the need for quantifying their impact. This is crucial for identifying the source term, optimizing Occupational Radiation Exposure (ORE), managing waste, and defining maintenance plans for ITER and, subsequently, for DEMO reactor and future Fusion Power Plants.
The EUROfusion consortium, within Work Package Preparation of ITER Operation (WPPrIO) is carrying out activities related to the development of computational tools for ACPs and experiments in a dedicated water-cooling loop under 14 MeV neutron irradiation at the Frascati Neutron Generator (FNG) and corrosion tests on CuCrZr ITER samples in baking conditions at RINA-CSM laboratories. The primary objective is to validate the OSCAR-Fusion code and other ACPs tools for evaluation of ACPs in conditions relevant to ITER, aiming to reduce uncertainties in ITER assessment. In this framework, an Engineering Grant (EEG) has been initiated to address the most challenging issues related to the impact of the ACPs to ITER ORE. The main aims of this EEG are to improve the reliability of the predictions and to develop multidisciplinary and advanced skills in ACP assessment for fusion applications.
This work provides an overview of the current status of these activities. Previous efforts are recalled, and some emphasis is placed on the activity in progress. Advancements include significant updates on the design of the water loop at FNG and on the relevant tests to be conducted as main part of the validation experiment. Additionally, preliminary results of corrosion tests, underway in the RINA-CSM laboratories, are presented and discussed.

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Submitted by **NOCE, Simone** on **Friday, January 26, 2024**

Abstract ID : **59**

Nuclear analyses for ITER Diagnostics Equatorial Ports

Content

Nuclear analyses are fundamental to support ITER port integration design and shielding optimization. In this framework, over the last years, ENEA has conducted detailed nuclear assessments on three equatorial diagnostic ports at different design stages. In particular, the analyses were performed in support of the Preliminary Design Review (PDR) of Equatorial Port #2 (EP#2), recently concluded, and are ongoing for the Final Design Review (FDR) of Equatorial Port #8 (EP #8), both of which host the Disruption Mitigation System (DMS). In particular, EP#8 presents unique challenges due to its proximity to the Neutral Beam Injection (NBI) sector, thus necessitating a model implementation over a tokamak sector larger than the usual 40°. As for the diagnostic port, EP #12, which has already undergone the FDR and is assumed as the a "standard" diagnostic port, comprehensive analyses are ongoing for the Manufacturing Readiness Review (MRR). The study includes modeling and calculations during and at the end of plasma operations from Port Plug to the Port Interspace in ITER 40° C-model, as well as in Port Cell integrated in a sector extracted from tokamak complex model using secondary sources for neutrons, and prompt and decay photons. All the analyses were conducted with D1SUNED v3.1.4 code and include the calculation of neutron and gamma fluxes, nuclear heating, radiation damage, gas and tritium production and shutdown dose rate at 12 days or at 1 day after shutdown. The impact of radiation cross-talk with neighboring ports around EP #12 was assessed along with the breakdown of the contribution of the main components to the total dose rate. Shielding studies and sensitivity analyses were also performed aimed at supporting design optimization and to mitigate the radiation levels in maintenance areas.

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Abstract ID : **60**

Advanced Shielding Materials for Spherical Tokamaks

Content

The recent progress in high temperature superconducting (HTS) magnets has significantly enhanced the potential of spherical tokamaks (ST) for harnessing fusion energy. Given the compact size of ST and high plasma density, the central column shielding (CCS) is subjected to neutronics and thermomechanical challenges such as high neutron load, volumetric heating from neutrons and gamma radiation, thermal load from the plasma, and structural loads from the shield itself. The limited space available for the CCS emphasizes the critical importance of selecting shielding materials for heat removal, long magnet lifespan, and low radioactive waste. The chosen material must exhibit exceptional neutron and gamma attenuation while minimizing radioactive waste production during operation, and good mechanical, and thermal properties.

In this study, we assess different potential shield candidates i.e., Tungsten, Borides, Carbides, and Hydrides in terms of their neutronics and thermomechanical responses. We performed neutron transport simulations with MCNP code for a 1GW fusion power ST, focusing on assessing the neutron load on various components, notably the CCS and HTS magnets. Through an analysis of neutron load, volumetric heating, and radioactive waste generation, we examine the suitability of different materials for CCS including the shielding and structural materials, choice of coolant, and their respective advancements. Our neutronics assessment points out the impact of alloying elements i.e., cobalt in tungsten carbide on the radioactive waste production. The thermomechanical responses and commercial availability of these potential materials are being explored. We also present the preliminary FEA modelling for the volumetric heating, impact of coolant and cooling channels on the effectiveness of CCS. Additionally, we compare these neutron loads with those observed in an ITER or DEMO like reactor, highlighting distinct material requirements for the spherical tokamak program.

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Submitted by **RAJPUT, Mayank** on **Friday, January 26, 2024**

Abstract ID : **61**

External libraries for D1S calculations for TRIPOLI-4 and MCNP-5/6 Monte Carlo codes in fusion applications

Content

The dose rate assessment is of utmost importance in fusion reactors because of the maintenance operations that these facilities will require. The reference way to perform this assessment for ITER nuclear analyses is the use of the D1S (Direct 1 Step) methodology, which consists in performing the neutron transport simulation, the material activation and the decay photon transport simulation in one step and thus in one Monte Carlo simulation only.

The current implementations of D1S are based on the MCNP-5 or the MCNP-6 Monte Carlo codes and require to have both the original source files and a patch file to apply on the sources to add the specific D1S functionalities, like for the D1SUNED tools 1. They also rely on specific libraries that implement the delayed activation gamma production in place of the prompt gamma production. Some important functionalities are added to the original MCNP codes to take into account the possible changes in the calculation configuration between the neutron transport phase and the photon one. The neutron phase corresponds to the plasma pulse, whereas the photon transport phase corresponds to a delayed phase where the activated radionuclides emit decay gammas. Useful additional functionalities also allow calculating the breakdown of the dose rate contribution per parent nuclide or daughter nuclide.

This work proposes to take benefit of the main options of the existing D1S codes (D1SUNED 1, Advanced-D1S [2]) and to implement them in external libraries. Instead of reading the so-called time factors that correct the decay gamma intensity to take into account the irradiation and decay phases in an external file by the D1S codes, they are included into the external library for a specific scenario, by correcting the decay gamma intensities with the time factors. Of course, the drawback of this implementation is that it requires having a library per irradiation/decay scenario. However, the calculations generally consider a well-known standard scenario like the SA2 one [3] in ITER (the conservative Safety Scenario for the experimental programme). Thus, the modification of the source file is not necessary and the compilation of the code, which can be a complex process on High-Performance Computing machines, is not required any more. The external libraries include:
* ENDF files where the prompt gamma production is replaced by the delayed gamma production
* ENDF files where the gamma production is totally removed for the nuclides that are not considered as being activated in the configuration to analyse
* ENDF files where the neutron transport cross sections are set to zero in order to mimic a material which is absent in the neutron transport, but present in the photon one (a detector inserted in the configuration after the irradiation for a dose rate measurement for instance)
* ENDF files where the photon transport cross sections are set to zero in order to mimic a material which is absent in the photon transport, but present in the neutron one (drained water in cooling pipes for instance after the plasma shutdown)

This study was applied first on the TRIPOLI-4® [4] Monte Carlo code for the sake of simplicity, as TRIPOLI-4® is the reference code for particle transport at CEA. The verification process relies on a comparison to a reference R2S (standing for Rigorous 2 Steps) methodology available in the TRIPOLI-4® code [5]. The verification analysis studied both neutron and photon responses in a simple configuration to check if the simulation was performing normally in a physical way. Then, a comparison with an experimental configuration in the FNG facility at Frascati was performed, the FNG Dose experiment [6]. The simulations are in good agreement with the experimental results.

Finally, the work proposes also to use the external ENDF libraries for the MCNP code. The NJOY-2016 [7] code allowed processing the ENDF file to build nuclear data libraries in ACE format for MCNP. Some tests with the MCNP code show that the use of the external libraries is possible for

other codes than TRIPOLI-4®.
The work still requires improvements in order to take into account for all the isotopes of interest the activation cross sections like the EAF-2010 ones and to focus on some specific points like the possible impact of the metastable state of some isotopes in the activation process, or some specific decay processes.

KEYWORDS: Dose rate, Monte Carlo, D1S, Fusion, Nuclear data

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Abstract ID : **62**

Compact sealed-tube D-D neutron generator with dedicated monitoring system suitable for ITER in situ calibration of neutron diagnostics

Content

Control of the D-D neutron source yield and energy distribution in case of accelerator-based sources is a tedious and important task. Minimization of uncertainty during ITER in situ calibration of neutron diagnostics including but not limited to Divertor Neutron Flux Monitor (DNFM) and Neutron Activation System requires a solution to this task in the form of a monitoring system. The monitoring system based on boron counters and neutron spectrometers – LaCl3:Ce and paraterphenyl is designed and tested with careful consideration of the realistic D-D neutron source characteristics corresponding to the VNIIA NG-24 model operation at nominal parameters of Iion = 2 mA and Vacc = 220 kV. One of the parts of this study is dedicated to the long-term stability and anisotropy study of the NG.
With GEANT4 software we model the detector responses and count-rate levels using the results of the neutron transport calculations with realistic source definition. The assessment is done for nominal operation parameters of the NG. We propose the optimal sensor location, dimensions and time resolution suitable for continuous expositions of ˜ 8 hrs at a time.
In the scope of this study we conducted testing of the monitor prototypes at several types of D-D neutron sources including ING-07D (Yn = ˜1e7 n/s), NG-14 (˜2e8 n/s) and NG-24 (up to 1.2e9 n/s). The calibration strategy for the monitoring system is discussed. Knowledge of ratio p/ β˜0.75 for LaCl3:Ce-based detector allows self-calibration of the energy axis of the monitor using its intrinsic radiation, demonstrating 5% accuracy of <En> reconstruction in experiment with the D-D NG. This setup including the neutron generator with neutron monitors was then used to study performance and verify sensitivity values for each section of the DNFM prototype fission chambers.
Additionally, we demonstrate the optimized monitoring system I&C architecture suitable for D-D NG that features preamplifier location close to boron counters and straightforward digitizing of neutron spectrometer signals.
As a result, we demonstrate the sufficiency of the monitoring system based on 2 Cd-shielded boron counters – and 2 neutron spectrometers for assessment of the NG total neutron yield and angle-wise average energy with time resolution varying from 1 to 60 s.
This research, its findings and results are supported in part by the state contract №H.4a.241.19.23.1014 as of 18th of January 2023 and by the Implementing Agreement No. 1 (ITER ref. IO/21/CT/4300002349) between Institution ‘‘Project Center ITER’’ and the ITER International Fusion Energy Organization “Research and Detailed Design Development of the components needed for Neutron Generators suitable for ITER in-situ neutron calibration’’.

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Abstract ID : **63**

Advanced Neutronics techniques of Spherical Tokamaks

Content

Due to the short time scales for the deployment of some government and commercial spherical tokamak programs, greater computational demands are being placed on the radiation transport codes which calculate key parameters such as magnet heating, tritium production and radiation doses. These are key attributes which are being used for component design and the safe operation of fusion devices.
Here we present a comparison of three Monte Carlo codes: MCNP, GEANT, and OpenMC. These codes have been run using a relevant tokamak geometry and source. The codes were run with all possible geometry configurations, Direct Accelerated Geometry (DAG) Constructive solid geometry and unstructured mesh and surface based meshes (stl). The codes were run for a tokamak model created using python code and Free CAD.
The results show a comparison of the radiation parameters for the codes as well as operational parameters such as run times and code memory usage. These data will be used to help with the development and utilisation of Monte Carlo codes for use in the design and operation of a fusion power plant.
For an example tokamak, all codes and methods agree with MCNP CSG neutron flux within 10 % for the neutron flux results. For the gamma results, OpenMC and the two MCNP methods agree but Geant 4 on average is predicting 20% more.

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Abstract ID : **64**

DT fusion power assessments on a tokamak based on line of sight neutron spectroscopy measurements at JET and prospects for ITER

Content

In the last decade, single crystal diamond detectors have been extensively used at JET for neutron spectroscopy measurements along collimated lines of sight. Although diamonds can measure 2.5 MeV neutrons, their use is optimized for 14 MeV neutrons. This is due to the exploitation of the 12C(n-α)9Be nuclear reaction channel which results in a well-defined gaussian peak in the recorded energy spectrum. Beyond their use as 14 MeV neutron spectrometer, in the last two JET deuterium-tritium (DT) experimental campaigns, diamonds have been exploited as DT neutron yield monitor. Furthermore, they can spectrally separate 2.5 MeV and 14 MeV neutrons providing a challenging DT fusion power measurement in trace tritium plasmas, when the neutron contribution due to deuterium-deuterium fusion reactions is important.
Diamonds have been cross-calibrated with the standard neutron yield diagnostics at JET and demonstrated to be reliable over the whole DT campaigns. Results from the JET DT campaigns will be described. Prospect for the DT fusion power measurements for ITER with diamond detectors inside the High Resolution Neutron Spectrometer (HRNS) project will be presented.

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Submitted by **RIGAMONTI, Davide** on **Friday, January 26, 2024**

Abstract ID : **65**

Neutronics analysis of lithium-chloride, lithium-fluoride salt as a tritium breeder for fusion reactors

Content

The Liquid Immersion Blanket (LIB) design employing lithium-based molten salt is a tritium breeding concept that requires less structural material, lower cost, and lower tritium inventory compared to other breeding concepts (Ferry et al. 2023). Many of the proposed breeder salts such as FLiBe (LiF + BeF2) require the use of highly hazardous beryllium-based compounds to act as a neutron multiplier. Many beryllium compounds are highly hazardous, thus any system working with beryllium-based salts will require extensive safety measures. In this work, an alternative, beryllium free salt breeder is proposed, ClLiF (LiCl + LiF), which employs the Cl37(n,2n) reaction and the high lithium fraction to produce a high tritium breeding ratio (TBR), particularly when enriched in Cl-37. A simple toroidal model of a beryllium multiplier and ClLiF salt with a 14.1-MeV-neutron ring source was created using the OpenMC neutronics code. A parameter scan of the effects of breeder thickness, external multiplier thickness, salt composition, chlorine enrichment, and operating temperature on the tritium breeding ratio (TBR) and energy multiplication was calculated using this OpenMC model. An operational parameter space for ClLiF is identified and the viability of this operational space is discussed. Additionally, a 1-D tokamak reactor based on the ARC radial build (Sorbom et al. 2015) was modeled in OpenMC with a blanket composed of either ClLiF, FLiBe, or lead-lithium, and the TBR, energy multiplication, and neutron spectra were calculated for each blanket composition and compared. Additionally, neutron activation resulting from DT fusion neutrons (14.1 MeV) in the simple toroidal geometry was modeled and the activities and contact dose were calculated using OpenMC and FISPACT for pure and impure forms of ClLiF, FLiBe, and lead-lithium. Lastly, the results of a small-scale tritium breeding experiment using ClLiF salt performed by the LIBRA group at the Plasma Science and Fusion Center (PSFC) are discussed in the context of the simulation results.

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Abstract ID : **66**

Recent Progress on Neutronic Analysis for ITER Diagnostics procured by JADA

Content

Japan Domestic Agency (JADA) has been performing neutronic analysis to optimize the shielding design and cooling system of ITER diagnostic systems procured by JADA (Microfission Chamber system, Poloidal Polarimeter system, Edge Thomson Scattering system, IR Thermography system, Divertor Impurity Monitor system, and Lower port integration).
Those systems except for the Microfission chamber system are optical diagnostic systems. Those optical components such as mirrors and shutters for protection of first mirrors are installed in the equatorial (EQ) and/or Upper port. Even though the optical paths of these systems have a labyrinthine structure, they contribute to an increase in the shut-down dose rate (SDDR) in the interspace because they create an opening in the port plug. Therefore, the shielding design including the optimum structure of optical paths based on the neutronic analysis using the C-model and/or E-lite model is essential to effectively reduce SDDR in the interspace, especially in the maintenance area of the corridor of the interspace. Since the increase in SDDR is not only due to radiation of peripheral components by streaming neutrons due to optical paths but also to radiation of the optical components themselves, adding shielding materials around optical components with consideration of their impact on the increase of SDDR is also important. Recently, JADA has been working on optimizing the shielding design of the optical system of the Poloidal Polarimeter system and Edge Thomson Scattering system.
In this meeting, details of the analysis results in the region of the port plug and interspace are presented.

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Submitted by **ISHIKAWA, Masao** on **Monday, January 29, 2024**

Abstract ID : **67**

Proof of concept for the propagation of input data uncertainties

Content

In the context of nuclear analyses, various uncertainty sources are dealt with, particularly those related to input data characterization. A systematically conservative approach attempts to account for all uncertainties in such a way that the result already covers all of them, which has been the usual approach in ITER. However, results exceeding limits or requirements following this approach represent a likely and valueless situation. Looking for alternatives, a proof of concept is carried out and presented here: two methods of uncertainty quantification due to the input data uncertainty are considered and compared. On the one hand, the Total Monte Carlo method leading to the characterization of a Probability Density Function (PDF) of a measurand, a tally response. On the other hand, the analytical expression of uncertainty propagation obtained with the first-order terms of the Taylor expansion of the same tally response. Both of them allow to study the likelihood of exceeding target values and the impact of input data uncertainties.

These schemes are applied to a simple but still relevant MCNP spherical geometry. It consists of a 2-meter-thick concrete shell, emulating the Tokamak Complex thickest walls. The concrete density and hydrogen content correspond to best estimates instead of the traditional conservative values assumed in ITER. Beyond the concrete shell, an additional void shell is considered as cell-tally for biological dose due to neutrons. An isotropic point source of neutrons of 14.1 MeV is found at the center of the geometry.

In this study only the concrete density and the hydrogen mass fraction are considered as input data uncertainties. Both methods are then applied to obtain a best estimate result plus a quantified uncertainty. Last but not least, the results obtained are also compared with the ones obtained using a systematically conservative approach. This work is concieved to identify potential advantages for ITER nuclear analys in phasing out the usual conservative approach in benefit of a best estimate with uncertainty quantification.

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Abstract ID : **68**

KATANA- closed water activation loop at the JSI TRIGA

Content

Numerous computational analyses of the water activation process have been performed for ITER, but the understanding of cooling water as a radiation source is still inadequate due to a lack of experimental nuclear data and inconsistencies among major nuclear data libraries, inaccurate computational methods/codes considering time- and spatial dependent radiation sources (such as the flow of activated water in the cooling system), experimental facilities to validate the methodology and, most importantly, the lack of water activation experiments under fusion-relevant conditions. For this reason, a closed water activation loop, namely KATANA, has been built in the Jožef Stefan Institute (JSI) at the research reactor TRIGA, facilitating a well-defined and stable 6 MeV-7 MeV gamma ray and neutron source. Such a high-energy radiation facility enables various experiments based on water activation, which are essential to fill knowledge gaps and improve existing experimental nuclear data sets, study detector response to high-energy gamma rays, explore short-lived moving radiation sources, validate computational codes and methods, etc. The project has been running for more than 2 years. The detector systems have been tested and the irradiation facility has gone through the approval/licencing process. The construction work was completed in November 2023 and first experiments have been carried out. Two experimental campaigns are planned as part of the EUROfusion collaboration in 2024 and 2025 with a possibility for more campaigns, should the need arise.
The water loop will be used for validation of computer codes, relevant for ITER, for testing of shielding material and testing of radiation detectors. The aim of this work is to present the closed water loop and its relevance to the conditions of ITER. Furthermore, a replacement of the inner irradiation part is planned for the second phase of operation, with the features of the first wall of ITER being preserved as far as possible. In addition, a modification of the non-irradiated downstream components (decay tank, collectors, etc.) is also being considered.

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Submitted by **PUTHANVEETIL, SUBHASH** on **Monday, January 29, 2024**

Abstract ID : **69**

Estimation of Shutdown Dose Rates for Occupational Radiation Exposure and implementation of ALARA in ITER

Content

In ITER's current research plan, up to 3×1027 neutrons are expected to be produced. This level of neutron production leads to nuclear integration challenges that will also be encountered on any nuclear fusion reactor. Here we focus on radiation protection of workers, with the Occupational Radiological Exposure (ORE) as the challenge. During outages for maintenance or assembly, workers will intervene in environments where they may be subjected to radiation from components activated by neutrons. The ITER Preliminary Safety Report emphasised the importance of applying the ALARA principle in design to demonstrate a collective dose of 500 mSv/year on average. As increased design maturity permitted more detailed analysis, the need for further study and optimization applying the ALARA approach across all project design activities became evident.
In 2019, an exercise to estimate ORE commenced focusing on Tokamak. Thirteen areas, representative of every maintenance area within the facility, were selected. Shutdown Dose Rates (SDDR) were determined for all areas, considering the SA-2 irradiation scenario, and compared with ITER requirements. Where discrepancies were found, dose reduction measures were proposed, leading to a creditable ORE estimation by 2022. This supported the need to streamline the ITER scientific program towards a two-stage approach, DT1 and DT2, to mitigate risks in licensing and project operation. Recently, SDDR calculations for the same areas were performed using the DT1 scenario, aiding decision-making for this re-baselining. We present the comparative results obtained with SA-2 and DT1 scenarios focusing on SDDR.
From a methodological standpoint, this work has introduced two interesting innovations. The definition of the activation corrosion product source in 3D high-definition resolution, along with more detailed source term determination, has been crucial in understanding this major contributing source to ORE. This includes its accumulation during Fusion Operation phases and the impact of baking. Additionally, the adoption of the ITER full-model MCNP model has proven to be a successful approach for comprehensive studies across the entire facility. These aspects will be explained in detail, and future prospects will be discussed.

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Abstract ID : **70**

Overview of UNED research activities in support of ITER neutronics

Content

The UNED research team has significantly contributed to ITER neutronics over the past decade. A noteworthy contribution in the area of computational tools is D1SUNED, which has become the primary tool for most neutronics studies due to its enhanced analysis capabilities and computational efficiency. Leveraging this tool, a significant milestone was achieved in geometric modelling: E-lite, the 360º heterogeneous representation of the Tokamak. These advancements have been crucial for conducting two important analyses: maintaining the ITER radiation atlas during machine operation, shutdown, and refurbishment (modes 0, 1, and 2), and the determination of shutdown dose rates in specific areas of the facility to produce consistent Occupational Radiation Exposure assessments in support of the ALARA (As Low As Reasonably Achievable) strategy.
Simultaneously, UNED was actively pursuing research lines to support ITER neutronics, with a growing degree of readiness. Tools like FLUNED, and more significantly GEOUNED, are being successfully deployed nowadays. The MCNP ITER full-model, which includes a description of the entire facility for the first time, is another notable example.
Moreover, UNED is exploring more ambitious and emerging research areas in four different aspects of computational neutronics. Firstly, research on the architecture of MCNP models is underway to reduce the computational demands of simulations, considering the extensive modelling introduced by E-lite and the Full-model. A neural network-based approach to variance reduction, which emphasizes sampling relevant events, has been successfully tested. Three practical implementations for propagating input data uncertainties to safety evaluation have been explored and compared under pertinent ITER conditions to integrate the systematic uncertainty quantification into the workflow. Lastly, a neural network-assisted, CAD-based, real-time dose estimator has been implemented in Spaceclaim to significantly speed up design tasks when photon sources are dominant; Note these include high relevance topics such as the evolution of the Tokamak Cooling Water System's 16N source, the design of the Hot Cell, and the implementation of the ALARA strategy.

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Submitted by **ALGUACIL, Javier** on **Monday, January 29, 2024**

Abstract ID : **73**

Comparative assessment of available toolsets for ITER nuclear analysis

Content

Nuclear analyses are one of the main activities needed in the design and optimization of the ITER machine. Because of the strong production of neutrons during the plasma discharge, the materials undergo activation, resulting in further release of gamma rays following the subsequent radioactive decay. The typical geometrical features of ITER and the need to precisely assess relevant quantities such as the activation, the shutdown dose rate, and neutron and photon fluence have motivated the development of complex models which are the input of neutron and photon transport simulations, generally based on Monte Carlo solvers. The neutronic solvers employed originate mainly from the needs of the fission calculations and are often the result of a giant effort spanning over decades. Consequently, the user interfaces and software architectures are not specifically suited for fusion applications. These considerations have motivated up to now several efforts to create tools and wrappers to facilitate the daily work of the analysts. After a brief introduction to the typical workflow employed in nuclear analysis, the presentation will deepen the analysis and comparison of the available toolsets.
More specifically, the ITER components are modeled through CAD (Computer-Aided Design) software which employs B-Rep (Boundary REPresentation), while Monte Carlo solvers typically employ CSG (Constructive Solid Geometries). The principal CAD library available as free and open-source software is OpenCascade, written in C++. Several projects are based on this library and its application to the nuclear industry has been already put in place for at least two decades. Still, its complexity led to the development of Python interfaces, which provide several levels of maturity and that may be a basis for further development of tools to support nuclear analysis. This presentation aims to illustrate the benefit of employing a programmatic approach in dealing with CAD processing and evaluate the maturity of the available toolkits.
From the point of view of the actual MCNP (Monte Carlo N-Particle) input decks, a plethora of projects are freely available, each one with a different philosophy and implementation. While recognizing the significant complexity imposed by the actual content of large and complex models, the common denominator of these projects is to assist the analyst during the development and in the course of actual nuclear assessment. At the same time, having a more friendly user interface may support verification, reinforcing the requirements of the safety reports. Starting from the analysis of a collection of these tools, the presentation will propose a personal point of view of the hypothetical ideal needs and architecture of the instruments useful to assist fusion neutronic activities.
Finally, the two worlds of the original B-Rep and the corresponding CSG should be connected, establishing a methodology to convert one to the other. This step represents one of the main time-consuming activities in charge of the nuclear analysts. Several approaches and codes are available, each one covering parts of this procedure. An assessment of the tools available will thus be proposed, taking into consideration the issue of the need to convert among potential different formats. Indeed, the methodologies to perform the visualization of such models will be treated as well, dealing with the potential concerns of being able from one side to visually inspect the model and from the other side to understand how a given transport solver behaves in a given region.

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Submitted by **PUTHANVEETIL, SUBHASH** on **Monday, January 29, 2024**

Abstract ID : **74**

Nuclear Heating Deposition in the ITER Vacuum Vessel

Content

This study addresses the critical concern of nuclear heat deposition (NHD) in the ITER Vacuum Vessel (VV) arising from plasma neutrons and prompt gammas. Comprehensive Monte Carlo radiation transport simulations were conducted, utilizing, for the first time, the 360º neutronics model of the tokamak, E lite.
A major update of E-lite was produced to estimate NHD values in the VV based on the most recent machine configuration. E lite_R220131 includes improvements in blanket representation by, on the one hand, implementing more recent models for blankets in rows #7-#12 with an explicit representation of the cooling water system, and, on the other hand, by introducing cooling water channels in the remaining blankets, including those in the irregular sector. Additionally, several missing, or incorrectly represented, components are implemented, such as correction coils, in-cryostat TCWS-IBED and TCWS-VV-PHTS, TDFS, TS shrouds, in-cryostat magnet feeders, and CCP in LP #7 and #13. Some issues identified in the NBI region are also fixed.
D1SUNED radiation transport code was employed in the NHD calculations. The study employed partial conservative estimates of integral heating, considering systematic modeling effects and random uncertainties. Furthermore, detailed 3D NHD maps were generated using the cell-under-voxel capability of D1SUNED, facilitating a material-specific assessment within the VV.
The analysis revealed an upper estimate for the vacuum vessel partial conservative heat deposition at 18.87 MW, indicating a 4.4% increment from the previous analysis. The nuclear heat deposition in the vacuum vessel inner shell exceeded the 0.61 W/cm3 criterion in numerous regions, primarily in poloidal sector #3 and triangular support of all toroidal sectors.
Differences with earlier work stemmed from factors such as heterogeneous blanket representation and new models for blankets in specific rows. The conservative estimate for power deposition during a 500 MW inductive DT pulse was 23.1 MW, increasing to 32.3 MW for the 700 MW pulse scenario.

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Submitted by **PUTHANVEETIL, SUBHASH** on **Thursday, February 1, 2024**

Abstract ID : **75**

14 MeV Neutron Source Facility at Institute for Plasma Research: Characteristics and Applications

Content

Institute for Plasma Research has developed a 14 MeV neutron generator facility which can produce neutrons of 1012 neutrons/second [1, 2]. The source is made of a deuterium ion accelerator and solid tritium target. The facility is commissioned successfully and it is operational now. The facility can be utilized for various application in the radiation field including the fusion technology. The facility can provide the maximum flux of 1010 n/cm2/s near to the target. The neutron source has been characterised with various diagnostics and yield measurement has been done using he associated alpha particle detector and foil activation [3]. This facility would be utilized for laboratory-scale experiments in the field of fusion reactor blankets, electronic equipment testing and radiography. Small scales design validation experiments of tritium breeding blanket as well as shield blanket can be performed. Activation characteristics of specific materials developed for fusion reactor can be tested. The facility can also be a premier destination for huge energy spectrum of neutron from thermal to 14 MeV. This feature opens the door for testing of electronics placed in ITER interspace, port cells and shielding corners. The irradiation experiments to study the degradation of magnet insulators and magnet conductors can be performed. Simulated results for various applications and experiments will be discussed in details along with capability of the neutron generator facility. The presentation will give a glimpse of the IPR neutron source facility features, fusion neutronics applications and probable utilization for ITER.

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Submitted by **PUTHANVEETIL, SUBHASH** on **Friday, February 2, 2024**

Abstract ID : **76**

Digital Twin Modelling Framework for Fusion Reactor Components

Content

The development of through-life management of commercial fusion power plants presents a unique set of challenges associated with the inherent complexities of these systems. Hence, the development of reliability and maintenance models for reactor components is one of the major challenges encountered in the realisation of a nuclear fusion power plant based on the tokamak concept. Furthermore, as fusion technologies are in their early phase of development, operational data from these systems are not widely available and their fabrication processes are not well industrialised, unlike other complex systems.
A digital twin can be developed to replicate many of the fusion reactor components by integrating models, deriving data and interpreting results to provide a holistic, digital representation. Since the concept of a digital twin can span many types of a thermonuclear fusion device, a focus has been placed on tokamak reactors and, more specifically, case studies will consider some of the most important plasma-facing components. These are studied as points of most concern due to the complex interaction of difficult to model plasma and non-standard materials. In short, the areas most likely to experience catastrophic failure.
An example component that could benefit from a high-fidelity digital twin is the "divertor"; playing a key role in extracting heat and ash from the fusion reaction, therefore minimising plasma contamination and protecting the surrounding walls from thermal and neutronic loads. The divertor is subject to very harsh conditions in normal operation with anticipated loads on the order of thousands of times the nominal levels during extreme events.
This paper attempts to review former and current digital twins envisioned within tokamak fusion reactors and thereby develop a hybrid modelling framework, with a series of coupled multi-scale, multi-physics computational models. An important part of this process will be verification, validation and uncertainty qualification (VVUQ) via available experimental data. Overall, this framework allows users to start developing digital twin models of in-vessel components. Key considerations are: to understand the plasma load on given materials; the potential failure modes; the modification of constituent materials; development of diagnostics through physical measurement and inference (via simulation codes or machine learning), as well as providing configurations, capabilities and limitations of testing methods and equipment. The aim is to improve confidence levels for future in-service monitoring and prognostics for predictive maintenance.

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Submitted by **PUTHANVEETIL, SUBHASH** on **Monday, February 5, 2024**

Abstract ID : **77**

SPARC Neutronics Models Applied to Neutron Flux Monitor Design

Content

SPARC is a compact, high field, tokamak under construction in Devens, MA, designed to operate with DT fuel and produce 14.1 MeV neutrons at rates up to 5e19 n/s 1. The present work demonstrates the capabilities of the neutronics models of the SPARC tokamak through application to the preliminary design of the neutron flux monitoring (NFM) system, one of four neutron diagnostics planned for SPARC early campaigns [2]. NFM will be calibrated to measure the real-time neutron yield rate (n/s) of the plasma, an input to the fusion power signal. NFM detectors of the cylindrical, gas filled, ion chamber style are located outside of the vessel in the tokamak hall. These detectors respond to either fast or thermal neutrons due to the choice of sensitive material, U238 or B10, respectively.

The design evaluation parameters for the NFM are dynamic range, time resolution, flat response over the energy spectrum of plasma emitted neutrons, response ratio of uncollided to total neutrons, and the time constant with respect to changes in plasma emissivity. The design levers are detector size and sensitivity, detector location in the tokamak hall, and the size, shape, and choice of flux shaping material local to the detector, used to shift the incident neutron energy or to attenuate select neutrons. The design tools are organized around two independent teams using particle transport codes MCNP [3] and OpenMC [4] with SPARC models developed using CAD conversion to unstructured mesh [5] and reductive CAD to constructive solid geometry. Important detector materials are modeled and simulations produce detector response in native units of counts per second or amperes, equivalent to the uncalibrated neutron yield rate signal. Finally, the design space also includes scenarios pertinent to commissioning and operations, where detector response is evaluated for neutron sources consistent with planned in situ point source calibrations and planned plasma reference discharges.

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Submitted by **PUTHANVEETIL, SUBHASH** on **Wednesday, February 7, 2024**

Abstract ID : **78**

Roadmap for Activated Corrosion Products (ACPs) Assessment in Fusion Reactors: A Global Collaborative Approach

Content

As the world transitions towards sustainable energy sources, fusion reactors emerge as a promising solution, harnessing the power of nuclear fusion to generate clean and abundant energy. The operation of fusion facilities will introduce challenges, one of which is due to the generation of Activated Corrosion Products (ACPs). ACPs are formed because of neutron activation and corrosion processes within the water primary cooling systems, necessitating a comprehensive methodology for their assessment. Currently, ITER ORGANISATION (IO) is using OSCAR-Fusion code for the ACPs assessment in the Tokamak Cooling Water System. In the frame of the safety demonstration activities, it is paramount to consolidate key input data, including activation rates and corrosion laws, used in ACPs studies.
The roadmap begins with a fundamental understanding of ACPs formation, emphasizing the need for a standardized methodology to assess their impact in terms of both investment protection and safety. Activation rates, influenced by neutron flux and irradiation time, play a crucial role in determining ACPs characteristics. The roadmap aims to consolidate data on activation rates through collaborative efforts, ensuring a comprehensive and accurate understanding applicable to diverse machine architectures.
Corrosion laws, governing the degradation of materials in the reactor environment, form another pivotal aspect of the roadmap. Different materials exhibit varying susceptibility to corrosion, and the roadmap advocates for a unified approach in collecting corrosion data. This involves collaborative research to establish corrosion laws applicable to fusion facilities conditions, facilitating a more precise assessment of corrosion impact on structural integrity, overall corrosion products inventory and its distribution within the cooling loops.
Global collaboration is the cornerstone of this roadmap, recognizing that a collective effort is essential to address the multifaceted challenges associated with ACPs assessment. Involving international stakeholders, including researchers, scientists, and engineers, ensures a diverse pool of expertise and resources. The roadmap envisions the establishment of a collaborative and synergetic R&D program where stakeholders contribute knowledge, share data, and collectively validate the proposed methodology. Establishing a platform for regular communication, knowledge exchange, and collaborative research will enhance the efficiency of ACPs assessment methodologies. This collaborative framework will not only accelerate the validation process but also promote a deeper understanding of ACPs behaviour in fusion applications.
Furthermore, the roadmap advocates for the establishment of standardized protocols for data collection and assessment. By harmonizing methodologies, international stakeholders can ensure consistency in results, enabling a more accurate comparison and interpretation of ACPs data. This standardization will contribute to the development of a universally accepted framework for ACPs assessment in fusion reactors.
In conclusion, the roadmap for ACPs assessment in fusion reactors presents a comprehensive approach to validate the methodology by consolidating key input data and fostering global collaboration, ultimately contributing to the safe and efficient operation of fusion facilities on a global scale.

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Submitted by **PUTHANVEETIL, SUBHASH** on **Monday, February 12, 2024**

Abstract ID : **79**

Setup of recent neutronic experiments in support of ITER at JET and at FNG

Content

The EUROfusion activities on the technological exploitation of deuterium-tritium campaigns at JET (DTE2 and DTE3), started in the frame of work package (WP) JET3 and currently under PrIO (Preparation of ITER Operations), aimed at exploiting the ITER-relevant radiation fields to improve the knowledge of nuclear technology and safety and to develop and validate nuclear codes, data and experimental techniques through dedicated and unique experiments. Among them, the validation of the numerical tools used for shutdown dose rate (SDDR) calculation for ITER through the comparison between numerical predictions and measurements with a dosimetry system based on ion chambers, the on-line measurement of tritium production in the mock-up of the ITER HCPB-TBM (Helium Cooled Pebble Bed - Test Blanket Module) with a diamond detector and the first-time experiment to measure the neutron-induced activation of cooling water during DTE3 campaign in the JET basement for the validation of the main computational tools developed for such important assessment for worker radiation exposure. Staying on the same subject, as a significant source of radiological hazard in ITER is due to the Activated Corrosion Products (ACPs) circulating in the cooling system, an experiment is currently under design at the 14-MeV Frascati Neutron Generator (FNG) to prove the accuracy of the ITER reference code in this matter (i.e., OSCAR-Fusion) under fusion relevant conditions, as this code was mainly validated in fission environment. In line with the contents of the present work, considering how the study of Single Event Effects (SEE) on electronics is crucial for ITER, it is worth mentioning the development of a modular irradiation station at ENEA named GENeuSIS (General Experimental Neutron System Irradiation Station), to be installed at FNG and which aims at reproducing specific neutron and gamma energy spectra for studying the response of devices and diagnostics.
The scope of the present work is to give to the audience of experts an overview on the experimental setup of the mentioned experiments, to describe their status and criticalities and to discuss the optimization of the next steps.

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Abstract ID : **80**

Application of high throughput neutronics simulation in fusion power plant design framework Bluemira

Content

To improve the concept design process of fusion power plants, neutronics simulations can be used to provide insight into in-vessel component design. Bluemira is an integrated inter-disciplinary design tool for future fusion reactors. It incorporates several modules to carry out a range of typical conceptual fusion reactor design activities. In order to improve the design of first wall in fusion reactors, one would need to run neutronics simulations to obtain the relevant quantities, including the tritium breeding ratio (TBR), heat load, neutron damage (dpa/fpy), and energy multiplication factor. The initial design point can then be further optimized as part of the first-wall design procedure using an iterative approach.
The length of time required by a high-fidelity neutronics simulation is prohibitively long for a high throughput procedure such as bluemira. Therefore we have created a pipeline to automatically create a lower fidelity CAD model of the in-vessel components of the tokamak, which can be fed into OpenMC, reducing the runtime down to the order of 10s, allowing for quick iterations to improve the first-wall design.

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