AGENDA

XIth ITER Neutronics Meeting

23rd - 27th May, 2016
Karlsruhe · Germany

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Remote participation is enabled through the web based Adobe Connect conference service provided by the German Research Network DFN ("Deutsches Forschungsnetz"). It works on any platform with any browser.

To join the meeting click on one of the links given below for each day. The VC screen will open in your default browser. You need to enter your name to identify yourself to the other VC participants. No password or e-mail address is required to enter the meeting room.

Monday, May 23, 2016
Meeting Name: ITER n-Mtg Day 1
To join the meeting:
https://webconf.vc.dfn.de/r3qcfksmxuw/

Tuesday, May 24, 2016
Meeting Name: ITER n-Mtg Day 2
To join the meeting:
https://webconf.vc.dfn.de/r86x4wkj5rz/

Wednesday, May 25, 2016
Meeting Name: ITER n-Mtg Day 3
To join the meeting:
https://webconf.vc.dfn.de/r421jkgm4ab/

Thursday, May 26, 2016
Meeting Name: ITER n-Mtg Day 4
To join the meeting:
https://webconf.vc.dfn.de/r1o9eawvcl3/

Friday, May 27, 2016
Meeting Name: ITER n-Mtg Day 5
To join the meeting:
https://webconf.vc.dfn.de/r1420v52sf9/

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ABSTRACTS

DAY 1
Introduction to KIT and to KIT´s Fusion Program

K. Hesch

Karlsruhe Institute of Technology, Fusion Program Management, Hermann-v.-Helmholtz-Platz 1, 76344 Eggenstein-Leopoldshafen, Germany

Email corresponding author: klaus.hesch@kit.edu

KIT is the research university in the Helmholtz Association of German Research Centres. It has evolved out of the merger of the Karlsruhe Technical University and the Karlsruhe Research Centre. It has more than 9,000 employees and more than 25,000 students, and is focusing on the challenges about energy, information, and mobility.

The mission of the KIT Nuclear Fusion Program is the development of technologies and materials for fusion energy. It involves ~ 220 scientists, engineers and support staff from 8 KIT institutes, and is working along three major lines, i.e.:

(i) Design, engineering, realisation and testing of components and systems for ITER
(ii) Critical developments towards DEMO and a Fusion Power Plant
(iii) Contributions to Wendelstein 7-X

There is a broad thematic coverage, ranging from superconducting fusion magnets and magnet components, through microwave sources for plasma heating & current drive, and port & plant engineering including the development of port plugs and launcher systems, remote maintenance, plant logistics, plant system & dynamics as well as balance of plant, to the deuterium-tritium fuel cycle, the in-vessel-components (breeding blanket, divertor) and the related structural and functional materials. For the materials as well as for the systems and components development, the prediction of neutronics behaviour based upon the development of neutronics methods and the improvement and validation of neutronics data, is a cross-cutting, key competence of the program.
Nuclear Integration at ITER

M J Loughlin\textsuperscript{1}, V Barabash\textsuperscript{1}, M Dentan\textsuperscript{1}, S Jakhar\textsuperscript{1}, G Janeschitz\textsuperscript{1}, D Leichtle\textsuperscript{2}, R Pampin\textsuperscript{2}, E Polunovskiy\textsuperscript{1}, L Sexton\textsuperscript{1}, A Tchistiakov\textsuperscript{1}

\textsuperscript{1}ITER Organization, Building 72/1118, CIO, Central Integration Office
Route de Vinon-sur-Verdon - CS 90 046 - 13067 St Paul Lez Durance Cedex – France

\textsuperscript{2}Fusion for Energy, Josep Pla, Torres Diagonal Litoral B3, 08019 Barcelona, Spain

Email corresponding author: michael.loughlin@iter.org

The Nuclear Integration Unit, an organisational trans-departmental entity inside IO-CT and at the same time a joint IO-CT/DA ITER team has been established to co-ordinate all nuclear integration activities. It is devised to ensure a consistent and fully integrated nuclear engineering approach throughout the whole ITER project compliant with project and regulatory requirements.

It will deal with all issues related to harsh radiation conditions for equipment and potential occupational radiation exposure to workers and internal exposure hazards are associated with contamination spreading of tritium and activated dust. These complex issues have a wide impact on the design, construction, operation, maintenance and safety of the ITER facility.

Nuclear integration is a complex transversal function which involves multiple systems and interfaces. A close interaction with the involved structures, systems or components design activities is required to ensure that respective requirements for shielding, radiation exposure for plant and personnel, material and equipment design and selection, and minimisation of radioactive contamination can be met in the integrated system. As regards limitation of exposure to ionising radiation the processes have to provide demonstration of compliance with ALARA principles.

This presentation will describe the objectives, the structure and the priorities of the Unit and the strategy to fulfil these objectives.
Abstract #3

XI\textsuperscript{th} ITER Neutronics Meeting, Karlsruhe, Germany 23 May – 27 May 2016

Overview of ENEA neutronics

R. Villari\textsuperscript{1}, M. Angelone\textsuperscript{1}, P. Batistoni\textsuperscript{1}, D. Flammini\textsuperscript{1}, N. Fonnesu\textsuperscript{1}, S. Loreti\textsuperscript{1}, F. Moro\textsuperscript{1}, M. Pillon\textsuperscript{1}, R. Pilotti\textsuperscript{1}

\textsuperscript{1}ENEA, FSN Department, ENEA C. R. Frascati, via E. Fermi 45, 00044 Frascati (Roma), Italy

Email corresponding author: rosaria.villari@enea.it

The present talk provides an overview of the recent ITER relevant neutronics activities carried-out in ENEA Frascati.

Support to C-lite development through the integration of semi-detailed blanket modules of rows 7-12 and 17&18 and nuclear analyses have been carried-out with MCNP5 for In-Vessel Viewing System (IVVS) port cell and Radial Neutron Camera (RNC) within F4E Contracts. The shutdown dose rate has been assessed with Advanced D1S method to verify the design limits. The contribution of the activated water to the nuclear loads in port cell area has been studied too. Furthermore studies on neutron produced through photonuclear reactions due to runaway electrons have been conducted.

Several computational and experimental activities have been performed in preparation for the future DT campaign (DTE-2) at JET for Streaming and Shutdown Dose rate experiment for the validation of ITER relevant codes, as well on activation of ITER relevant materials and calibration of DT generator in the frame of EUROfusion WPJET3 project.

Further nuclear analyses and improvements of Advanced D1S code for reactor studies have been carried-out for DEMO design in the frame the EUROfusion Power Plant Physics & Technology Programme (PPPT).

Concerning experimental activities, progresses were achieved in developing neutron and tritium monitors for TBM within the Framework Partnership Agreement (FPA-395) between the “Consortium for Nuclear Data Studies/Experiments in Support of TBM Activities” and F4E. Positive tests of diamond neutron detectors up to 240°C and of the prototype self-powered neutron detector with Cr emitter under intense gamma field and also at high temperature (up to 450°C) were performed.

Neutronics activities devoted to the conceptual design of the “New Sorgentina” Fusion Source (NSFS) project and the Divertor Tokamak Test (DTT) facility have been carried-out as well.
Supported by the projects related to ITER and fusion reactors, neutronics technologies in China have been greatly improved in the recent years, including the development of simulation codes, nuclear data library, neutronics facilities & experiments, and nuclear analyses.

The latest version of SuperMC, the advanced Monte Carlo(MC) neutronics code featured to deal with complicated nuclear systems can accomplish the transport calculation of n, γ, and is integrated with the functions of automatic modeling, visualized analysis and cloud computing. Several advanced acceleration method were developed, which speeds up the global flux map calculation 249 times for the ITER A-Lite model. The activation calculation function was under development. SuperMC has passed a series of ITER benchmarking calculation and been applied in ITER nuclear design and safety analysis, such as the shielding analysis of ITER bio-shield plug.

The High Intensity D-T Fusion Neutron Generator (HINEG) project has been launched. The R&D of HINEG includes two phases: HINEG-I and HINEG-II. HINEG-I, which is designed to generate both the steady beam and pulsed beam, has been completed and commissioning since the end of 2015 with the D-T fusion neutron yield of up to $10^{12}$ n/s. HINEG-II aims at a higher neutron yield of $10^{14}$~$10^{15}$ n/s. HINEG can be used for research of nuclear technology and safety including the validation of neutronics method and software, radiation shielding and protection, materials activation and radiation damage as well as neutronics performance of components.

Since 2003, FDS Team has participated in the neutronics studies of ITER and finished over ten ITER tasks covering the building and updating of ITER neutronics reference models, nuclear analysis for port plugs and activation analysis for cooling water, etc. In 2015, a new task of shielding analysis of Bio-shield plug in B1 of ITER Tokamak was finished, providing recommendations for the shielding design.

CN ITER TBM program, in which INEST•FDS Team undertake the safety and relevant neutronics analysis task. Under the support of CN MOST, a small TBM mock-up was fabricated for neutronics test purpose, and a DT neutron validation experiment has been finished. The preliminary experiment result will help to reduce the gap between theoretical and experiment.
An overview is presented on KIT’s recent activities in the field of fusion neutronics related to ITER. The activities include development works on computational tools and methods relevant to ITER neutronics, the modelling of specific systems, and related design and performance analyses conducted in the frame of ITER and F4E contracts.

Complementary development works of interest to ITER, performed in the frame of EUROfusion’s PPPT (Power Plant Physics and Technology) and JET projects as well as specific F4E grants, are also addressed. These include, among others, the development of specific nuclear data evaluations/libraries provided to the international community (and ITER) through the NEA Data Bank, Paris, and the IAEA, Vienna. Such data libraries include also co-variance data as required e.g. for uncertainty assessments of nuclear responses in ITER, other nuclear fusion facilities or experiments. A related application example is shown for the Monte Carlo based TBR uncertainty assessment of the European DEMO tokamak.
Abstract #6

XI\textsuperscript{th} ITER Neutronics Meeting, Karlsruhe, Germany 23 May – 27 May 2016

Overview of UNED neutronics activities for ITER

R. Juárez\textsuperscript{1,+}, J.P. Catalán\textsuperscript{1}, P. Sauvan\textsuperscript{1}, F. Ogando\textsuperscript{1}, A.J. López\textsuperscript{1}, A. Kolsek\textsuperscript{1}, R. García\textsuperscript{1}, M. García\textsuperscript{1}, J. Sanz\textsuperscript{1}

\textsuperscript{1} UNED, ETSII Calle Juan del Rosal 12, Madrid, Spain

+Email corresponding author: rjuarez@ind.uned.es

UNED neutronics research team is involved in different activities related to ITER, and the progress since X\textsuperscript{th} ITER Neutronics Meeting of the most relevant ones is given here.

For the ITER Diagnostics Division, UNED has been working, among others, in the following main topics: i) Shutdown Dose Rates reduction in the diagnostics equatorial ports, ii) Generation and updating of an MCNP model of the Diagnostics Generic Equatorial Port Plug (DGEPP), and iii) MCNP code modification to improve the radiation source modelling in the port cells.

For the ITER Analysis Section division, UNED has been working in two main tasks: i) Coils heating due to decay water in an alternative in-vessel cooling pipes arrangement under PCR-662, and ii) Development of the MCNP Tokamak Building reference model, a 360\textdegree realistic radiation source for ITER operation, and production of updated radiation maps.

UNED has been also involved in methodology improvements interesting for ITER neutronics. A prominent R2S-UNED new capability is shown: a methodology to propagate the decay gamma source uncertainties to the SDDR.

In addition, D1S-UNED new capabilities are also presented here:

- Analysis: i) computation of decay gamma source uncertainty, ii) generation of the spatial distribution of the source regions contribution to a SDDR tally
- Acceleration techniques based in the decay gammas population control
- R2S related capabilities: i) generation of decay gamma sources compatible with Common Decay Gama Source (CDGS) format, ii) reading and gamma transport of decay gamma sources, including source portability iii) propagation of the decay gamma source uncertainty to the SDDR.
Abstract #7

XI\textsuperscript{th} ITER Neutronics Meeting, Karlsruhe, Germany 23 May – 27 May 2016

Development and Application of ITER Fusion Neutronics Capabilities at Oak Ridge National Laboratory

R. Grove\textsuperscript{1}, S. Wilson\textsuperscript{1}, K. Royston\textsuperscript{1}, A. Ibrahim\textsuperscript{1}, J. Risner\textsuperscript{1}, S. Mosher\textsuperscript{1}, S. Johnson\textsuperscript{1}

\textsuperscript{1}Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA

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The radiation transport computations required for the ITER neutronics community to provide analyses that support robust design solutions are among the largest, most challenging neutronics simulations being performed and they stretch existing capabilities. This presentation will provide an overview of activities in the past year within the Radiation Transport (RT) group of the Oak Ridge National Laboratory (USA) to perform ITER neutronics analyses and to develop, apply and disseminate neutronics tools that are focused on making robust design solutions for ITER possible and practical. We will briefly describe the analyses we have performed and collaborated to perform for ITER, describe the current suite of advanced tools we use, discuss the challenges and gaps that we perceive, and present our plans for next generation capabilities.

The RT group develops and applies computational methods and software to provide analyses and design solutions for a broad range of radiation transport applications areas including shielding, radiation effects, dosimetry/measurement, radiation protection, fusion neutronics, reactor physics, criticality safety, accelerator and beamline design, global security, non-proliferation, isotope production, and sensitivity/uncertainty. A key characteristic is the combination of expertise in design, analysis, methods, and software development that spans this broad range of applications. We develop and apply state-of-the-art Monte Carlo, deterministic, and hybrid radiation transport computational tools as well as sensitivity analysis, uncertainty quantification and optimization tools for analysis of complex problems of international significance. Methods/software tools developed and applied in these areas are designed to be compatible with and complementary to each other in order to provide flexible capability for a wide range of applications and computer architectures. They are designed to solve the largest, most challenging problems on high performance supercomputer architectures and to be scalable down to laptop computers.
Neutronics analysis is essential for the safe design and operation of the ITER experimental facility. Traditionally performed using MCNP, the creation of constructive solid geometry (CSG) models for neutronics calculations requires extensive effort and expert judgment by the analyst to simplify and convert CAD models whilst retaining transport-critical features.

As the design of ITER systems progresses to maturity, increasing levels of geometric detail will be required (as with the in-vessel region), and the scope of the modelling will increase (larger machine sector, inclusion of neighbouring systems, transport into port cells). This will require increased analyst effort to create the models, and increased computational requirements to run them.

New developments in radiation transport codes have the potential to address some of these issues. The use of an unstructured surface or volume mesh should reduce the analyst time needed to create the geometry and clean models, enabling a rapid turn-around of calculations. Neutronics analysis might then keep up with the engineering design cycle, and become more agile with models that can be quickly adapted to explore changing designs. A higher level of detail could also be accommodated with minimal additional analyst effort - the potential downside being increased computational requirements.

For a simplified tokamak model with ITER-like features, comparisons have been performed between MCNP6 (CSG) and alternative transport codes and techniques including the unstructured volume mesh capability of MCNP6, the unstructured surface mesh of DAG-MCNP5, and Serpent-2 (CSG, unstructured mesh and unstructured surface). Comparisons will be both qualitative and quantitative, showing tally results for neutron flux and heating in meshes and cells, computational requirements, and assessing capabilities and further development needs for fusion neutronics.

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Using PyNE to Extend the DAGMC Framework to ITER Analysis Workflows

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Nuclear analysis of complex shielding problems like ITER is increasingly relying on intricate workflows with multiple steps. The rigorous-2-step (R2S) approach to shutdown dose rates is one such workflow; most implementations require activation calculations on a detailed mesh between two Monte Carlo radiation transport steps. Hybrid Monte Carlo/deterministic methods are another example, with both deterministic and Monte Carlo radiation transport performed on the same model. In each case, when a CAD based geometric model is used directly for the Monte Carlo radiation transport steps, then tools are necessary to make that same model available for other steps.

The PyNE toolkit has been extended to support these workflows. Anchored in the method to discretize DAGMC geometries into Cartesian grids, additional tools are available to generate input for and process output from ALARA and PARTISN. PyNE also contains generic methods for sampling photon sources from Cartesian grids that support flexible biasing schemes.

As defined by the MS-CADIS concept\textsuperscript{1}, these two workflows can be combined in an effort to use deterministic results to improve the efficiency of shutdown dose rate analysis. The GT-CADIS implementation uses a linearization approximation of the activation operator to estimate effective shutdown photon production cross-sections for use in an MS-CADIS paradigm. This optimizes the Monte Carlo neutron transport in phase space for the ultimate photon dose rate problem.

This presentation will discuss the improvements to R2S and hybrid workflows and demonstrate and analyze some of the features of the GT-CADIS methodology based on the components available in the PyNE toolkit.

Radiation Transport in the Tokamak Complex To Assess the Impact of Activated Cooling Water

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Short-lived radioisotopes of nitrogen (N-16 and N-17) are created in the IBED system due to activation of oxygen in the cooling water by high-energy neutrons from the plasma. The N-16 decays primarily by emission of high-energy gamma rays and the N-17 decays primarily by emission of intermediate-energy neutrons. This analysis determined the spatial distribution of N-16 gamma and N-17 neutron responses from IBED activated water in the tokamak complex on levels L3 and L4, in the vicinity of the divertor port cell, and outside the tokamak complex due to skyshine following the completion of PCR-662 updates. This work also served as an independent assessment of the analysis in ITER_D_QZ7BEK\textsuperscript{2} - Analysis of Radiation Transport Due to Activated Tokamak Cooling Water. Advanced variance reduction techniques were used with Monte Carlo methods to determine radiation responses on a fine spatial mesh with a high degree of precision. The response maps generated from the radiation fields are used to characterize the biological dose rate, accumulated dose in silicon, 1-MeV silicon equivalent neutron fluence, and the total neutron flux due to N-16 and N-17 sources in the IBED activated coolant. Biological dose rates were compared with RPrS radiation zoning requirements, while the remaining responses are necessary for determining the placement of critical and non-critical electronics in the tokamak complex. In addition, the results of a recent scoping analysis of changes to the total neutron flux due to moving activated coolant inside the bioshield will be discussed. Advanced visualization tools used to facilitate model checking and to post-process transport simulation results will also be presented.

\textsuperscript{2}S. Jakhar, “Nuclear Analysis of 16N and 17N Radiation Fields from IBED Activated Water During Plasma Operations, “ ID\textsuperscript{M} UID – QZ7BEK
The effect of $\text{B}_4\text{C}$ on the SDR around the ITER device

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Two configurations of the ITER device were modelled: one baseline case and one case with the addition of a layer of $\text{B}_4\text{C}$. The baseline case includes the standard C-lite CAD model, a detailed model of the Cryopump, generic equatorial port plug including interspace frame and equipment, and detailed upper port with interspace frame and equipment. Neutron transport was performed using DAG-MCNP5v1.6 and used a DAGMC-enabled version of ADVANTG to provide variance reduction parameters. Pyne.r2s was used to couple the DAGMC geometry and neutron results to perform a R2S-based calculation of the shutdown dose rate, using the SA2 irradiation scenario and determining the shutdown sources at $10^5,10^6,10^7$ s. It was found that the addition of the $\text{B}_4\text{C}$ layer reduced the SDR by a factor of two near the bioshield with efficacy being reduced proportionally further away. Furthermore, at every decay time the SDR was not below the 100 $\mu$Sv/hr guideline. It was also determined that the crosstalk in most cases was significant, especially in the case of the upper port where crosstalk from the PFC2 region was significant.
Radiation Transport through the Neutral Beam Injection System Beyond the Tokamak Complex

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The magnitude and spatial distribution of the biological dose rate response due to radiation transport from tokamak plasma through the neutral beam injection (NBI) system and out of the tokamak complex has been characterized. The biological dose rate outside the tokamak complex due to water activated within the NBI coolant system has also been characterized. The total biological dose rate exiting the north wall of the ITER tokamak complex due to transport through the NBI system was of particular interest and results have been examined relative to the goal limit of 0.44 $\mu$Sv/hr outside the building. The biological dose rate due to the plasma neutron source was analysed in Subtask 2\textsuperscript{3} of Task Agreement C74TD21FU. Subtask 4\textsuperscript{4} of the same task agreement characterized the source due to decay of N-16 in activated water in the IBED and NBI primary heat transfer systems and calculated the resulting biological dose rate outside the north wall of the tokamak complex. Skyshine effects were included in both analyses using a site model extending from the tokamak complex to the administrative building north of the nuclear platform. Both sets of calculations employed Monte Carlo splices and hybrid variance reduction methods via the ADVANTG code. A review of results, methods and computational challenges encountered in these analyses will be presented.


\textsuperscript{4} K. Royston and S. Wilson, “Analysis of Radiation Transport from the Neutral Beam Cell Due to Activated Cooling Water,” US_D_23C2VC
ABSTRACTS

DAY 2
Nuclear Analysis of the Torus Dust Filter System

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During ITER operation radioactive and tritiated metallic dust will be formed, principally through erosion of plasma facing materials. This metallic dust can pose a radiological hazard during pump down of the vacuum vessel as some will be evacuated along with the gases. In order to arrest these metallic dust particles a Torus Dust Filtering System (TDFS) is incorporated into the torus roughing line. The system comprises of four High Efficiency Particulate Absorption (HEPA) filter sets. These are positioned in the interspaces of upper ports 4, 5, 6 and 7.

Although the filter sets are designed to last the life-time of ITER and their replacement is not planned, the design allows for the remote maintenance/replacement for providing access to the upper port plug or in the event of an accident. Therefore hands-on access for quick checks or installation of some remote handling tools should be foreseen. Once removed hands-on access within the hot cell will be required for surveys of the vacuum valves, inspection of connectors etc.

In order to estimate the dose rates in the areas around the torus dust filters nuclear analysis has been performed. On load dose assessments were carried out to estimate the dose rate to sensitive components. It was found that the maximum combined on-load dose rate to silicon in a valve actuator would be 24.5 Gy/h. It was also shown that the dominant active nuclides in main components of the filter assembly were $^{51}$Cr, $^{60}$Co, $^{182}$Ta and $^{55}$Fe. Using the SA2 irradiation scenario dose rates during shutdown were also calculated including both dose from the activated dust itself and the activation of filter set. The dose rates from 10 kg of tungsten dust $10^6$ s after shutdown was estimated to be less than 100 µSv/h when a shield was applied. With the current design the shutdown dose rate 30 cm away due to the activation of the filter set only was estimated to be in the region of 1700 µSv/h.

This work was funded by the ITER organisation under contract IO/CT/15/4300001130. The views and opinions expressed do not necessarily reflect those of the ITER organisation which is not liable for any use that may be made of the information contained herein.
Due to the large complexity of the ITER device detailed computational analysis is difficult even with the largest of computational clusters. Methods have been devised to create simplified analysis, but there are large discrepancies on how accurate these analyses have been. For the nuclear analysis, material homogenization has been used, but has been shown to give inappropriate results very quickly through the ITER blanket shield modules. Also, changes in port region gaps and other geometric anomalies have caused problems when confirming nuclear results. So a method to be able to solve for very large complex models, while maintaining the appropriate level of detail, and be able to make changes quickly must be devised.

PPPL has been in the process of using the Attila neutronics code to create a method of taking sectioning of the ITER device and creating secondary calculations that take the solution of a previous calculation. This allows the work of different regions to be disjoint and calculated separately. Changes to the ITER device can be taken separately and not have to be calculated extensively over a large computational model.

The focus of this work is within the port cell. The nuclear calculation up to the bioshield pushes the current PPPL computational resources. Therefore a calculation was done up to the bioshield, then this solution is used to create a source spectrum to be used for the port cell analysis. This will allow the contribution of the plasma added to a calculation that incorporates the water activation and other sources. Giving a detailed analysis that does not require extensive computation, allowing multiple iterations of different configurations to be testes quickly. This work covers the code development and the results of the port cell nuclear analysis.
Shielding Analysis of Bio-shield Plug in B1 of ITER

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Bio-shield plugs play important roles in protecting electronic devices and workers during ITER operation and maintenance. Recently, a new bio-shield plug model was designed by DIN in order to propose a standardization of configuration and respecting the requirement. In this study, shielding analysis of bio-shield plugs in B1 has been performed with SuperMC and compared with MCNP.

Four typical bio-shield plugs, including Cryostat Cryopump plug, Torus Cryopump plug, IVVS plug, Remote Handling plug, and two reference plugs of ordinary concrete and heavy concrete were analyzed. A detailed 3D neutronics model of the bio-shield plugs in B1 including the penetrations have been developed from the original CAD model and material description. The shielding capability of each type of module of bio-shield plug has been investigated through the study of flux attenuation and material activation with a neutron source near to the bio-shield plug. Results shown the attenuation of neutron flux in IVVS plug was the slowest and the activation of structural materials in Torus Cryopump plug was the most serious. Then detailed analysis for Torus Cryopump and Remote Handling plugs was conducted by integrating the local model of plugs into the Tokamak building. It was found that for Torus Cryopump plug, the shutdown dose rate in port cells is about 20 times higher than the limit. The present design of bio-shield penetrated by Torus Cryopump does not provide enough radiation attenuation for the safety of workers and devices. For Remote Handling plug, the detailed neutron flux distribution in B1 was calculated and it was found the neutron flux has a reduction of about 4 orders of magnitude along the Port Cell longitudinal axis.

The comparing calculations which were given in the above analysis showed good agreement among the results of SuperMC and MCNP, and demonstrated the correctness and effectiveness of SuperMC for fusion applications. SuperMC has significant advantages and capabilities to model complex geometry which is the major obstacle of fusion neutronics analysis. SuperMC can be well applied for fusion neutronics analysis.
TRIPOLI-4® rigorous two-step scheme for shutdown dose rate calculation in fusion applications

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TRIPOLI-4® is a three-dimensional and continuous energy Monte Carlo particle transport code, developed by CEA, and dedicated to shielding, reactor physics with depletion, criticality safety and nuclear instrumentation. TRIPOLI-4® is currently able to simulate four kinds of particles: neutrons, photons, electrons and positrons.

An activation scheme has been developed in TRIPOLI-4® so as to calculate shutdown dose rates for fusion applications. This activation scheme should be used next year for ITER calculations performed by CEA.

This activation scheme is based on the Rigorous Two-Step (R2S) approach:

- The TRIPOLI-4® Monte Carlo code performs a neutron transport calculation in order to compute the flux in each region susceptible to produce decay gamma. The MENDEL depletion code (developed at CEA) computes the decay photon sources for each region based on the neutron fluxes previously calculated by TRIPOLI-4®.
- The TRIPOLI-4® code transports the decay photons and computes the dose rates induced in each region of interest.

In this talk, we present first the TRIPOLI-4® and the MENDEL codes, and the new activation scheme implemented in TRIPOLI-4® for the shutdown dose rates calculations for fusion applications. Then, we describe the TRIPOLI-4® developments and new capabilities recently realized in that aim. We present the results of a very simple user case performed to run an activation calculation. Finally, we identify the main developments needed in TRIPOLI-4® to improve this activation scheme.
One of the most active topics in the current ITER design related to the neutronics is the determination of the Shutdown Dose Rates (SDDR) and fulfilment of the different ITER related requirements; like 100 µSv/h after $10^6$ seconds of cooling time in the port interspace for planned in-situ maintenance operations.

The determination of SDDR is a complex problems answered with increasing sophistication in terms of tools to compute it. One of the tools currently available for SDDR determination is D1S-UNED, firstly presented in X$^{th}$ ITER Neutronics meeting last year. Since then, new capabilities have been implemented and are presented here:

- Analysis capabilities: i) decay gamma source error determination, ii) spatial distribution of the decay source contribution to a tally
- Acceleration techniques: i) decay gamma emission switch by universes and cells, ii) gamma emission per collision control by universe and cell
- R2S related capabilities: i) generation of decay gamma sources compatible with Common Decay Gama Source (CDGS) forma, ii) reading and gamma transport decay gamma sources, including source portability iii) propagation of the decay gamma source uncertainty to the SDDR.

A comparison of R2S and D1S methodologies is made also to quantify three uncertainties sources introduced by R2S systems and avoided by D1S systems: i) the neutron flux discretization in a mesh, ii) the neutron flux spectrum energy binning, and iii) the decay gamma emission spectrum binning. By comparing R2S-UNED and D1S-UNED answers in simple cases, as long as in one ITER representative case, an explicit evaluation of these three uncertainties source is given with relevancy to ITER.
'ACTYS'
An Activation Analysis Code For Fusion System

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Here, we present the details of newly developed indigenous nuclear activation solver called ACTYS. Rigorous nuclear activation calculations form an important element of all the three major phases of nuclear fusion facilities viz construction, operation and decommissioning. A suitable activation code must be able to treat all fusion relevant reactions and operational scenarios yielding accurate results with faster performance. ACTYS is developed as part of a long-term plan of developing a 3-D nuclear activation code that is able to yield results with good accuracy while maintaining faster performance when coupled to any complicated 3D fusion reactor geometry. ACTYS in its current form is able to perform point-activation calculations using well-known linear chain solver technique with improvements for long-lived chains. It has inbuilt data pre-processing modules that can read cross-section data from both ACE as well as ENDF formatted activation data libraries. PREPRO is used to particular to read ENDF files. Additional code modules are developed for condensation of activation data, into any arbitrary group structure. Post-processing of activation data yields nuclear quantities such as radioactivity levels, inventories with detailed pathways and parameters such as contact dose, decay heat, gamma source spectrum etc. Extensive validations have been performed using strategies such as validations against analytical calculations (decay-test), realistic irradiation tests which are relevant to ITER, and tests according to specific guideline proposed by IAEA Nuclear Data Section. FISPACT-2007, which is at present used for ITER activation calculations widely, is used for comparison with ACTYS in all test cases. The validations are chosen so as to cover almost all fusion-neutron induced nuclear activation reactions. Properly pulsed long enough activation time scenarios for fusion relevant materials are included in the validation process. In summary, ACTYS is found to satisfy all the necessary and sufficient conditions required for a fusion activation solver and expected as a stepping stone towards development of a 3-D activation solver.
Abstract #19

XI\textsuperscript{th} ITER Neutronics Meeting, Karlsruhe, Germany 23 May – 27 May 2016

Diagnostics Generic Equatorial Port Plug and shielding proposals for interspace

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ITER Diagnostic Division has developed an MCNP model of the Diagnostics Generic Port Plug and associated interspace components up to the bioshield plug with different aims: 1) To support the analysts and save resources avoiding work duplication, 2) To ensure configuration management and quality assurance of common components, 3) To standardize the modelling approach. This DGEPP model has been inserted into C-lite v2, and it has been distributed on demand.

Based on that MCNP, studies on different fronts to reduce Shutdown Dose Rates in the port interspaces after $10^6$ seconds of cooling time have been carried out:

- **VV inserts and rear shims.** Sensibility of SDDR in the interspace to the materials B\textsubscript{4}C, SS and W assigned to these two components has been investigated in 9 combinations

- **Impurities limitation.** It is known that Co impurities activation is one of the SDDR driver of for the interspace. A study on how different Co content in different components could reduce SDDR is made to support further Co content limitation.

- **Shielding proposals outside the port plug.** The VV port extension covering, Port Duct external covering, Port Duct inner covering, are explored considering different thicknesses and materials, like B\textsubscript{4}C, borated SS and SS. In addition, a shielding cabin placed in the Interspace Support Structure corridors only during maintenance and made of SS with different thickness is explored.

In this work the DGEPP and SDDR studies carried out with it are presented.
Radiation In-Port Cross-Talks of Diagnostics inside the ITER Equatorial and Upper Ports

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In many cases ITER Diagnostics Port Plugs host several Diagnostic systems. On examples of two Equatorial Port Plugs (EPP) #17 and #8, and components of Upper Port Plug (UPP) #3 it is shown the necessity to consider the radiation impacts between adjacent systems. This subject is important for Diagnostics designing at the stage of port integration to ensure engineering and maintenance solutions for the Diagnostic tenant systems.

Multiple sets of diagnostic equipment inserted inside the same Port Plug create additional pathways for radiation streaming along the diagnostic channels and labyrinths (e.g. optical pathways). The entanglement of the pathways impacts on key neutronics aspects of the Diagnostics measuring itself such as readings of neutron spectra by the Tangential Neutron Spectrometer (TNS) in EPP#8, or possibility for maintenance access to the Diagnostics from the Port Interspace (PI) area quantified in terms of Shut-Down Dose Rate (SDDR). We present several interesting effects of in-port cross-talks, such as the gamma shadow effect of the Tritium and Deposition Monitor (TDM) shield block. This TDM block reduces the SDDR inside the PI of EPP#17 originated by radioactive sources of Core-Imaging X-ray Spectrometer (CIXS). The TDM apertures are compensated by the proper arrangement of its shield block in PI. The methodology principles of the SDDR analyses with use of parallel MCNP transport interfacing with activation code FISPACT through the mesh-based R2Smesh or D1S codes are described in work⁵. The CIXS inside the Diagnostic Generic EPP⁶ is analysed in EPP#17 local model, while EPP#8 is modelled globally with C-lite v2. This talk demonstrated that in order to take advantage of particular shielding improvements in full extent, we should also assess the mutual influence of every diagnostic system installed inside the same port.

IFN/ESS-Bilbao work in Preliminary neutronics calculations for ITER Diagnostic ports

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In the context of Diagnostic ports design, one of the activities to be performed is the neutron transport and activation analysis. This analysis delivers several results: It produces a heat generation map for mechanical calculations, it calculates neutronic damage and Helium generation in First Wall, and neutron activation through the entire port. This activation calculation is then used to calculate a gamma source over the activated materials than will later be used for shutdown dose rate calculations. IFN/ESS-B (Instituto de Fusion Nuclear/ESS-Bilbao) performs these calculations as a subcontractor for IDOM ingenieria, and has developed an inhouse tool coupling MCNP5/6 and ACAB activation code. The MCNP modelling of the ports is described, along with the results for the first preliminary design for Equatorial Port 10 and Upper Port 01 are described. The results point out the need of further refining and the introduction of neutron capture material.
In this work we used the CAD based DAG-MCNP5v1.60 transport code to analyze detailed models inserted into a 40 degree partially homogenized ITER global model. The regions analyzed include the neutral beam injection (NBI) region and the pellet injector flight tubes. The NBI models contain a 40 degree NBI sector with complete rows of detailed BM13-BM16 CAD models inserted. One of the NBI models consists of sector 2 containing two heating neutral beam ports (HNB1, HNB2) as well as the diagnostic neutral beam (DNB) port. The other NBI model consists of sector 3 with the HNB3 port. Nuclear heating and DPA in the vacuum vessel (VV) was determined. For the sector 2 region, three versions of a DNB liner were examined to improve VV heating and dpa. For the pellet injector flight tubes, nuclear heating was determined for the tube supports as well as the tubes themselves. In all cases, nuclear heating mesh tally results were provided to be mapped to ANSYS finite element meshes for subsequent thermal analysis.
High-resolution heating distribution calculated for the ITER vacuum vessel with updated C-lite MCNP model

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Nuclear heating of the vacuum vessel (VV) is an important issue for the design and the safe operation of ITER. The heating distribution must be known with high accuracy to identify hot spots which may be crucial for the reliable operation. The VV is heated by neutrons passing through the blanket shield modules and gaps, and photons generated in the VV structure. The heating distribution is thus strongly affected by materials and geometry of in-vessel components. An accurate representation of these components is therefore a key prerequisite for reliable results of the nuclear heating distribution in the VV.

In the frame of an F4E Task Order C-lite, the Monte Carlo code MCNP reference model of ITER, was updated by AMEC, CCFE and KIT-ENEA with a semi-detailed representation of the in-vessel components (IVC) as currently designed for blanket rows 7 to 12 accordingly to work⁷. Semi-detailed IVC models in blanket rows 1 - 6 and 13 - 18 were already available, although corresponding to an earlier design stage.

The updated C-lite model was applied to compute distributions of the nuclear heating in the VV with the MCNP6 Monte Carlo code using mesh tallies with a resolution of 2 cm. The calculations were performed on the HELIOS supercomputer located in Rokkasho, Japan.

The presentation reports the results of the new heating calculations. The results confirm the VV hot spots obtained previously for C-lite with a simplified IVC representation but give less conservative results for VV regions behind the updated IVCs.

⁷ A. Turner et al, Updating of the In-vessel Components for the C-lite Model, Xth ITER Neutronics Meeting, Cadarache, France 30 June - 3 July 2015.
Peculiarities and Challenges of the ITER Test Blanket Systems Nuclear Analyses

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During 2010 – 2015 Fusion for Energy has executed several contracts on Test Blanket Systems (TBS) nuclear analyses. The implementation of the sequence and integration of design structural analyses and nuclear analyses iterations was an important achievement. The works objectives, methodology and results have been reported\textsuperscript{8,9,10} and presented at the conceptual design reviews. Complementary to the published papers, this presentation provides a systematic overview of the specifics of the TBS nuclear analyses and shares the challenges met and the experience gained in applying and adapting the ITER generic methodology. In general, TBS peculiarities are derived from the proximity of the Test Blanket Modules (TBM) to the plasma; presence of the port gaps; flowing LiPb; proximity to the lower port; and presence of activated LiPb in the port cell (PC). Among others the subjects discussed will include: adaptation of the ITER irradiation scenario to components with different lifetimes and flowing liquid metal; translation of complex CAD geometries into MCNP models; selection of shielding materials by scoping calculations; grouping of different activation sources, including those from activated corrosion products (ACP) and PbLi residues. We realized that the pipe bends inside the TBM shield are critical to limit the neutron and gamma streaming and explored the use of WC and B\textsubscript{4}C to reduce the doses in the interspace. The radiation level in PC behind the bioshield is defined by the source at bioshield and the PbLi activation. Recently, doses from PbLi spilt into PC after an accident have also been evaluated. Thus far the potential biological hazard for inhalation and ingestion were considered by indicating the inventories of the toxic, radioactively hazardous, long-lived and \textsuperscript{210}Po precursor radionuclides. The tritium extraction, permeation and outgassing of the activation inventories, activation of ancillary loops due to nitrogen decay neutron radiation and dust will be addressed in the future.

\textsuperscript{10} L. Packer, et al, “EU tritium breeding system shielding design for ITER and overview of candidate DEMO blanket concepts”, paper ENC2014-A0212 European Nuclear Conference, 11-14 May 2014, Marseille, France
ABSTRACTS

DAY 3
Abstract #25

XI\textsuperscript{th} ITER Neutronics Meeting, Karlsruhe, Germany 23 May – 27 May 2016

**Nuclear Analysis of the ITER Radial Neutron Camera architectural options**

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The ITER Radial Neutron Camera (RNC) is a multichannel detection system hosted in the Equatorial Port Plug 1 (EPP1) designed to provide information on the neutron source total strength and emissivity profiles through the measurement of the uncollided neutron flux along a set of collimated lines of sight (LOS). Furthermore, the ion temperature profile and fuel ratio ($n_d/n_t$) can be assessed by means of line-integrated neutron spectral measurement.

The RNC consists of two sub-systems based on a fan-shaped array of cylindrical collimators: the ex-port LOSs, covering the plasma core, embedded in a massive shielding block located in the Port Interspace, and the in-port LOSs distributed in two removable cassette integrated inside the Port Plug.

Presently, the RNC layout development process is undergoing a System Level Design phase: several preliminary architectural options based on a System Engineering approach have been defined for both the ex-port and in-port systems. A detailed nuclear analysis of these options has been performed through radiation transport calculations with the MCNP Monte Carlo code.

The MCNP model of each RNC architectural option has been developed and recursively integrated in an upgraded version of the ITER MCNP C-lite model where all the details of the EPP1 and nearby diagnostic systems have been included. Successively, the expected neutron spectra and the secondary photon background at the detectors positions have been evaluated. Moreover, the radiation environment in the EPP1 has been fully characterised through the assessment of 3D neutrons, gammas and decay gammas flux maps. Finally, the impact of a reduced ex-port shielding block on the neutron and gamma spectra has been investigated. The results of the present study provide guidelines for the development of the RNC final design and the necessary data for the measurement performance analysis.
Neutronics Analysis and Activation Calculation for X-Ray Crystal Spectrometer of ITER

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Neutronics and activation analysis have been carried out for X-ray Crystal Spectrometer (XRCS) sight tube, which will be installed in equatorial port no. 11 assigned for the ITER diagnostics. The neutron transport calculations are performed using Monte-Carlo N-Particle code (MCNP). The transport results are used for the design and optimization of a proper radiation shield for the sight tube. The base C-lite neutronics ITER model is grossly modified to include all required details of the port plug, diagnostic apertures and the diagnostic system. The sight tube is supposed to be placed in the interspace, after the closure plate, to channel the X-rays to the spectrometer. A complete radioactive inventory calculations along with contact doses and nuclear activity levels are obtained for two different kind of sight tube material. FISPACT-2007, an inventory code is used for this purpose. The analysis for this particular sight tube can be used to obtain a material preference based on radiation point of view. Further, the dependence of neutron spectrum and irradiation time on the activity levels, contact dose rate and the production of dominant dose contributing radio-nuclides have been studied. Dominant radionuclides which contributing up to 95\% of the total dose are identified and their pathways are generated to back trace their sources, as an effort to reduce the dose rate. The effect of reducing Cobalt content in SS-316 on SDDR is evaluated separately for the sight tube of the XRCS system. Many of the FISPACT calculations are repeated with ACTYS, a locally developed activation solver.
Preliminary Neutronics assessments of the Collective Thompson Scattering Diagnostics

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The Collective Thomson Scattering (CTS) diagnostic for ITER will probe variations in the alpha-particle velocity distribution by directing a 1MW 60GHz gyrotron beam into the plasma while monitoring the backscattered microwaves. Through the use of waveguides and mirrors these microwaves are transmitted to the diagnostics building for detection.

Large cut-outs of the ITER blanket are required to inject the gyrotron beam and to extract the signal, leading to challenges in providing sufficient shielding of the equatorial port plug #12 where the CTS diagnostic will be placed.

The diagnostics system is developed in a partnership between DTU and IST and has recently passed the System Level Design. Neutronics efforts focus on supporting the system design. In order to reduce computational efforts and allowing for parametric investigations a simplified MCNP model, called Torus, has been developed and benchmarked against the generic DGEPP model, which serves as our baseline.

The Torus model is applied in parameter studies of the CTS diagnostic and provides a useful aid for the engineering design. Examples are shown, including calculation of the shutdown dose-rates for various options of shielding in the region close to the port plug backplate.

In addition, a simplified geometry of the CTS-system capturing only the main features of the design has been implemented into the DGEPP model. The resulting fast neutron fluxes and shutdown dose rates in the region close to the backplate are benchmarked against earlier estimates calculated by IO.
Development and Application of Monte Carlo Program SuperMC for ITER Neutronics Analysis

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Due to the complexity of ITER on geometry and neutron physics, the Monte Carlo (MC) methods have been broadly adopted in ITER neutronics analysis. However, great challenges exist for the current MC methods including the modeling of complex geometries, deep penetration problem simulation etc. MC program SuperMC is evolving to better support the ITER Neutronics Analysis.

The latest version of SuperMC can accomplish the transport calculation of $n, \gamma$. It is integrated with the functions of automatic modeling, visualization and cloud computing. It has powerful automatic and accurate modeling of complex-configuration geometry and convenient physical modeling including fusion plasma source, material, tally etc. Facet based transport geometry has been implemented in SuperMC, which supports MC simulation directly using complicate ITER CAD models without pre-processing. To simulate the deep penetration problem in radiation shielding, a Global Weight Window Generator based on uniform particle density was proposed and developed, which is computationally efficient and requires little user interference. It speeds up the global flux calculation by 249 times for the ITER A-Lite model.

SuperMC has been verified and validated by more than 2000 benchmark models and experiments such as Shielding Integral Benchmark Archive Database (SINBAD). To validate the comprehensive capability for ITER neutronics analysis, SuperMC has passed the calculation of ITER Benchmark model, A-Lite and C-Lite models. With the distinguished capabilities for complex geometry modelling, SuperMC has supported to construct a series of ITER reference neutronics models which has been released to 14 nations. Furthermore, SuperMC has been successfully applied in ITER nuclear design and safety analysis, including the shielding analysis of ITER bio-shield plug, the maintenance plan assessment of PF coil, etc.
Towards the use of Geant4 for fusion neutronics analysis

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Geant4 is an open source toolkit for simulations of particles, including the neutron transport using evaluated cross-section data. To use it for fusion neutronics analyses, one important issue is the modeling of complex fusion reactor geometries in Geant4. In addition, obtaining the high fidelity spatial neutron distributions is also important for the detail data analysis. In this work, an advanced open source modeling system has been developed to model CAD and unstructured mesh geometries for Geant4.

The modeling of CAD geometry is achieved through the automatic MC geometry modeling program McCad, which is supporting MC codes MCNP and TRIPOLI as well. It converts the CAD geometry into a new type CSG solid called half-space solid. The Geometry Description Markup Language (GDML) format has been adopted and extended to support the new solid type. It is possible to combine this solid type with the tessellated solid type in one GDML file.

The modeling of unstructured meshes is achieved using tetrahedron, pyramid, wedge and hexahedron elements, which have been developed as new Geant4 solid types. A superimposed mesh scoring capability has been developed for obtaining spatial distributions of physical quantities overlaying the mesh on the actual geometry. Interfaces have been developed for importing unstructured meshes from files and export the scoring results into a file in Visualization Toolkit (VTK) format.

Comparisons have been made between Geant4 primitives, tessellated solid and the newly developed half-space solid types. The unstructured mesh scoring has been used to obtained agree results with that of the MCNP6 code in calculating a test case model. In addition, the ITER benchmark model has been adopted, and volumes of the solids calculated in the MC model agree well with that of the CAD model. Further test and verifications have to be carried out in order to conclude the reliability of the entire modeling system.
Materials Activity in Support of Nuclear Integration at ITER

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Materials activity is defined as a part of the scope of Nuclear Integration Unit and this activity is a part of fully integrated nuclear engineering approach throughout the whole ITER project compliant with project and regulatory requirements.

The main scope of the materials activity includes:

- Define and implement the radiation-relevant requirements for materials for the ITER components
- Define permissible materials, chemical composition, specified and non-specified impurities content taking into account ALARA principles for the engineering material selection and justification
- Provide reliable validated data needed for neutronic analysis (chemical composition, impurities, etc.) taking into account the status of on-going procurement and materials which already procured for various ITER components
- To define and co-ordinate experimental measurements to quantify the radiological significant impurities in materials
- To maintain a material sample archive need for the future detailed analysis
- To assess the application of non-standard materials for shielding application based on ALARA principles
- To define and co-ordinate experimental measurements to verify the predictions of radiation transport calculations for the ITER components

A close collaboration of materials activity needed for nuclear analysis and procurement activities will provide the back-ground for the reliable analyses needed for the safety assessment.

This presentation will describe the objectives, current and planned activities in materials area to support Nuclear Integration Unit.
ABSTRACTS
DAY 4
The development of the electron cyclotron heating upper launcher which is going to be installed in 4 of ITERs upper ports has made huge progress in recent years. In step with this development several MCNP models were devised from the CAD models to always calculate up-to-date neutronic results. A brief review of the ECHUL development, the current status and an outlook of the neutronic analyses for the electron cyclotron heating upper launcher is presented.

In the case of ITERs charge exchange recombination spectroscopy upper port plug the development passed the conceptual design review in 2015. Currently several design options are being investigated regarding optical performance, engineering feasibility, cost, maintenance especially with respect to remote handling and the performance of their neutron radiation shielding. The current status and outlook of this investigation from the neutronic performance point of view is presented.
Neutron production from de-confined runaway electrons in ITER

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The first phase of ITER is the hydrogen phase, a non-nuclear phase, planned mainly for full commissioning of the tokamak system in a non-nuclear environment where extensive remote handling is not required. However, there are at least two physical processes that will generate neutrons during this phase: the presence of impurities (Be) in the plasma, that can trigger nuclear reactions, and the bremsstrahlung photons, generated by the runaway electrons, which can produce photonuclear reactions. Despite the total amount of neutrons produced in this ways is negligible to the neutron yield foreseen during the DT or DD plasma phases, those neutrons could be relevant in some cases. In particular, the de-confined runaway electrons can strike almost at any point of the FW, with an energy that can be up to several MeV, in a much localized region of the wall. This will produce neutrons in a small region with energies up to that of the impinging electrons, generating hot spots of activation that need to be evaluated. Moreover, those neutrons can be used as a diagnostic for runaway electrons.

In this study Monte Carlo simulation with the MCNP6 code has been used to simulate the de-confined runaway electrons, as an electron beam impinging on the ITER plasma facing components. The goal of this study is to characterize the neutron emission induced by the runaways and to provide neutron sources to be used with MCNP for subsequent calculations. Different scenarios have been considered: interaction of electrons with a blanket module and with the divertor, as well as two different spectra for the runaway electrons, corresponding to two possible scenarios. The assumptions made for the description of runaway electrons are discussed and the neutron sources is generated and validated.
MCNP simulation of X-rays from Runaway Electrons and other directionally anisotropic sources with the ITER C-lite model

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In a tokamak runaway electrons (RE) can carry a substantial part of the plasma current. They can interact only with plasma – confined RE – and with the PFC/vessel structures of a tokamak – de-confined RE. In both cases hard x-rays are generated. The runaway electrons may lead to substantial damage of the first wall and other in-vessel components. In ITER, it is expected that the REs can gain energy as high as 100 MeV with currents up to few MAs.

The calculations of the x-rays induced by runaway electrons, as are expected in ITER, are presented. The fluxes and spectra at the locations of interest for the ITER X-ray diagnostics, are calculated with the MCNP6 code. The ITER C-lite model, covering 40° of the geometry, cannot directly be used for calculations with point sources, as is the de-confined RE source, nor for directionally oriented sources, as the confined RE source. To our knowledge no 360° model of ITER with a level of detail of the C-lite model exists, which would enable calculations of the described problems.

We have previously introduced a procedure for calculation with point sources. In the present work this is expanded to be used for directionally oriented sources, as is the source due to confined REs. The procedure is based on an alteration of the MCNP code, effectively tracking neutrons as if they were transported in the full 360° and not reflected by the reflecting surfaces of the 40° C-lite model. For directionally anisotropic sources, covering more than one sector, the procedure has to be expanded by the use of addition routines for correct treatment of the direction change in every other sector of the expanded C-lite model. The described procedures could possibly be used also for calculations, connected to the planned calibration of the ITER diagnostics systems.
Calculations to support JET neutron yield calibration: Model of a compact DT neutron generator

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A compact DT neutron generator was chosen as a calibration source for the upcoming calibration of the JET’s neutron detectors to 14 MeV neutrons. Unfortunately, DT neutron generators are complex neutron sources as the energy of emitted neutron depends on the angle of emission, on the beam energy, on the configuration of the generator itself, and, in some cases, on the presence of D/T mixtures in the beam and in the target. Their intensity, moreover, is not necessarily constant. A full characterization and calibration is therefore needed for the DT neutron generator selected for the neutron calibration at JET. As part of a complex strategy for the neutron generator characterization, diamond detectors were chosen to measure the neutron energy spectra at different angles. The technical characteristics of the selected generator mean that there are multiple source components as a result of the use of a mixed beam where both deuterium and tritium are present in the ion beam and the target. For this reason, the simulation of generated spectra has required the development of a suitable neutron source routine. Finally, the internal composition of the neutron generator is proprietary and was only disclosed to us in a form of a simplified sketch which was then translated into the MCNP model.

The characterisation of the generator was performed at the National Physical Laboratory (UK) as a collaboration of various institutes and laboratories that provided their state-of-the-art detectors and expertise. The results of these measurements will be used to improve the reproduction of the generator’s neutron emission in MCNP simulations.

The process of modelling a compact neutron generator into a calibration source of neutrons will be presented, current status explained and major challenges discussed.

* See the Appendix of F. Romanelli et al., Proceedings of the 25th IAEA Fusion Energy Conference 2014, Saint Petersburg, Russia
ADVANTG automates the process of generating the variance reduction parameters for continuous energy Monte Carlo simulations using MCNP5. It generates space and energy dependent mesh based weight window bounds and biased source distributions using 3D discrete ordinates solutions of the adjoint transport by Denovo.

SEA has performed the benchmarking of MCNPX code with NESDIP2 experiments which were performed at ASPIS facility of the NESTOR reactor at Winfrith, with the purpose of qualifying the calculation methodology and nuclear data used for neutron fluence calculation in vessel and cavity for improving reactor dosimetry of the pressure vessel of Light Water Reactors.

The computed results are presented in Table 1 showing a good match with measurements. A gain in computer time of 4 times is obtained and for the same number of particles a 27% lower uncertainty is obtained. In Figures the MCNP geometry model and the weight-window distribution for the detector in the cavity are shown.

### Table 1. Dosimetry results comparison

<table>
<thead>
<tr>
<th>Position and distance</th>
<th>Rh103(n,n')</th>
<th>In115(n,n')</th>
<th>S32(n,p)P32</th>
</tr>
</thead>
<tbody>
<tr>
<td>Water-0cm</td>
<td>1.02</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Water-1.6cm</td>
<td>0.99</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Water-6.6cm</td>
<td>0.95</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Water-10.5cm</td>
<td>0.91</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Water-12.1cm</td>
<td>0.93</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Water-18.0cm</td>
<td>1.03</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Water-20.7cm</td>
<td>0.98</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Water-23.5cm</td>
<td>0.99</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Water-28.5cm</td>
<td>0.97</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Water-31.2cm</td>
<td>1.04</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>RPV-3.4cm</td>
<td>0.97</td>
<td>1.06</td>
<td>1.00</td>
</tr>
<tr>
<td>RPV-39.7cm</td>
<td>1.02</td>
<td>1.07</td>
<td>1.06</td>
</tr>
<tr>
<td>RPV-45.4cm</td>
<td>1.04</td>
<td>1.07</td>
<td>1.06</td>
</tr>
<tr>
<td>RPV-51.1cm</td>
<td>1.06</td>
<td>1.06</td>
<td>1.02</td>
</tr>
<tr>
<td>Cavity-60.5cm</td>
<td>0.94</td>
<td>1.02</td>
<td>1.03</td>
</tr>
</tbody>
</table>
Analyses of ADVANTG input parameter variations on the NEXP streaming benchmark

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Neutron flux and dose calculations of the ITER machine far from the plasma source are challenging due to the complexity of the ITER geometries and the physical process involved. Experimental verification of the state-of-the-art of the computer transport codes and nuclear data which will be used for these calculations is therefore mandatory. NEXP benchmark is currently under preparation at JET with the objective to assess the performance of codes and nuclear data used in ITER neutron streaming calculations.

The Automated Variance Reduction Generator (ADVANTG) code¹¹ reported is a Monte Carlo/Deterministic Hybrid transport code developed by ORNL, using Denovo¹² and MCNP¹³. Its approach for combining deterministic and Monte Carlo transport methods is based on the Consistent Adjoint Driven Importance Sampling (CADIS)¹⁴ method. The fundamental concept is to generate an approximate importance function from a fast-running deterministic adjoint calculation and use the importance map to construct variance reduction parameters that can accelerate tally convergence in the Monte Carlo simulation.

Variations of the input parameters for ADVANTG have been performed, such as different multigroup data libraries (27n19g, 200n47g, DPLUS and BPLUS), spatial mesh voxel size, number of polar and azimuthal angles, order of expansion of the scattering anisotropy modeled and different methods for determining variance reduction parameters (CADIS, FW-CADIS). A comparison of the impact on FOM (Figure-of-merit) and other statistical test because of input parameter variations has been done. A significant impact on FOM when using different versions of MCNP (MCNP5 version 1.6 and MCNP6 version 1.0) has been observed.

*See the Appendix of F. Romanelli et al., Proceedings of the 25th IAEA Fusion Energy Conference 2014, Saint Petersburg, Russia

In-vessel modelling updates for C-lite

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This paper presents the updates to the in-vessel components in the C-lite model carried out by F4E, IO-CT, AMEC FW and CCFE. The work concerned all the blanket modules and the divertor, the outboard divertor rails, the divertor interfaces and the Lower Port-Torus Cryopump. In order to optimize the overall schedule, the tasks have been divided in the following way:

a) BM#1-6 according to CM Release 07/2015 and PCR-646/641 (by AMEC FW)
b) BM#7-8-9 according to PCR-709 and with the inclusion of FS (by F4E)
c) BM#10-11 according to PCR-709 and CM Release 02/2016 (by CCFE)
d) BM#12 according to CM Release 02/2016 regarding Manifold Attachment Shield (by F4E)
e) BM#13-18 according to CM Release 07/2015 (by AMEC FW)
f) Divertor Model according to DM Release 07/2015 (by AMEC FW)
g) Outboard Divertor Rails according to CM Release 02/2016 (by F4E)
h) LP-TCP according to CM Release 02/2016 and PCR-713 (by F4E)

Depending on the amount of the modification required, the MCNP models have been updated from the previous versions or newly created from the CM CAD. The DM information, important to neutronics responses, has been implemented as much as reasonably possible. In all cases, the corresponding simplified CAD has also been provided for the final C-lite release assembly. In addition to the standard F4E quality nuclear assurance programme, “Recommendations for In-Vessel Neutronics Modelling”¹⁵ have been developed in collaboration with IO-CT and provided to the stakeholders in order to homogenize the outcomes as much as possible.

All the models generated will be included in the 06/2016 C-lite Reference Model Release.

¹⁵ F4E_D_24CE99
Nuclear analyses for ITER Systems, Structures and Components (SSC) rely on faithful and accurate geometry models of the specific SSC but also, due to the nature of radiation transport in complex geometries, on adequate boundary conditions provided by global models. This is particularly relevant for the calculation of nuclear responses in or affected by extended areas of the problem space.

To this end, ITER Organisation provides reference nuclear models, including but not limited to the 40° tokamak torus sector up to the bioshield. A series of so-called Lite-models have been developed and released over the past several years, the last official one C-lite V1 as of October 2013. Since then several modelling activities over a wide range of systems have been performed to follow-up configuration and design changes. Several local versions of enhanced C-lite models have been used frequently.

A joint effort is currently devoted to establish a new official release model in June 2016 with a set on modelling, verification and user support documentation. Based on recent modifications of the block universe structure in the C-lite MCNP model to accommodate better the interfaces between SSC major model updates will be integrated. This includes most prominently Vacuum Vessel and In-vessel systems. The full content of the new reference model with generic diagnostic port systems and torus cryopump, Remote Handling and IVVS lower ports will be presented.

Upon finalisation and successful testing MCNP and CAD representations will be released to the ITER neutronics community. A corresponding system model database will need to be continuously populated to allow easy access, exchange and control of system specific models as needed and as required by dedicated nuclear analysis.
Progress on a ITER Nuclear Analysis Framework

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Nuclear analyses are key activities for ITER integrated design, which involves multiple sources, systems, interfaces and processes in a complex radiation environment. They are technically challenging and computationally demanding, and require specific tools and expertise for proper execution, monitoring and verification. Nuclear analysis activities are frequently classified as Protection Important Activities (PIA) due to their link with safety-relevant parameters (e.g. activity and doses) or the design of Protection Important Class components. Therefore they must be conducted in compliance with requirements of the INB Order for the safety demonstration, in particular regarding reliability, accuracy, appropriateness and qualification of data, models, methods and tools. In addition to the requirements linked with their PIA classification and the proper planning and deployment of qualified resources to provide the required analyses, there is also the general need to integrate and couple these activities to design processes of SSCs and to provide a complete, consistent and rigorous framework for the methodology and quality of ITER nuclear analysis.

To this end, the new ITER Nuclear Integration Unit has implemented this distinguished scope of “Quality of Nuclear Analysis” encompassing procedures, guidelines, references and standards to ensure technical quality, coherence with ITER baseline and compliance with INB prescriptions. This presentation will provide an overview of the related management plan, but will focus on high priority tasks. Among those are firstly a set of high-level QA procedures defining roles & responsibilities in planning, preparation, conduction, control and acceptance of ITER nuclear analyses. Also specific instructions and guidelines for shielding analyses (e.g. calculation of SDDR) and verification & validation instructions are currently prepared. Besides producing project documentation, to be integrated in the Quality Assurance Programme and into Configuration Management as appropriate, a continuous effort is required on nuclear models and analyses storage and maintenance, which relies on joint and collaborative efforts throughout the ITER project.
Abstract #40

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Facet-based Transport Simulation Method Directly Using CAD Models
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Accurately describing geometry of new nuclear facilities during design phase is one of the main difficulties in Monte Carlo (MC) based nuclear transport simulation for ITER. Various methods such as CAD to MC geometry conversion methods that could generate Constructive Solid Geometry (CSG) based MC geometry from CAD models and ray tracing method directly using CAD models have been developed to aid MC geometry describing. Currently, the conversion methods from CAD to CSG geometry needs a lot of manual pre-processing, which leads to low potential of modelling efficiency and needs improvement. Existing ray tracing methods directly on CAD models are relevantly slow in calculation speed, still the calculation accuracy is not revealed.

Lately, facet based ray tracing has been implemented in SuperMC, supporting generating facet based transport models in arbitrary accuracy and ray-tracing on the them. It adopted a two level Binary Space Partition (BSP) forest to accelerate the ray tracing, where the high level BSP tree split the space in solid level and the low level BSP tree split one solid in triangle level. With the BSP forest, each step in the ray tracing would involve few solids and few triangles for one solid, as a result would greatly reduce the calculation load.

Several tests were carried out based on ITER benchmark model, on which SuperMC has already been validated. Different calculation situations, such as flux on complex geometry, flux on small target after deep penetration was tested. The memory consumption, calculation speed and result accuracy were compared between CSG based method and the new method. The results demonstrated that our implementation of facet based MC geometry has achieved similar calculation efficiency as CSG geometry and with nearly zero geometry conversion costs, while keeps good agreement in the results as CSG geometry.

The new facet based geometry describing and ray-tracing methods has already been applied on a model with abundant high order spline surfaces, the Force Free Helical Reactor (FFHR) model which is a helical-type fusion reactor with key parts twisted. The calculation was successfully conducted on the original model without pre-processing, reveals great convenience with facet based ray tracing over CSG geometry.
Improved Solid Decomposition Algorithms for the CAD-to-MC Conversion Tool McCad

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McCad is a geometry conversion tool developed at the Karlsruhe Institute of Technology (KIT) for the automatic conversion of CAD models into the constructive solid geometry (CSG) representation. The resulting geometry models can then be used in Monte Carlo (MC) particle transport simulations applied in design analyses of fusion reactors such as ITER and the European DEMO. The conversion of such a CAD model necessitates decomposing complex solids into a collection of disjoint and simple convex solids. The decomposition algorithm implemented previously in McCad turned out to be not very efficient and robust when applied to large and complex geometry models resulting in frequent programme crashes, and irregular and fragmentized solids. Furthermore, a lot of CPU time and memory are consumed.

To overcome such difficulties and improve the capability of McCad, new decomposition algorithms and functions have been developed. These include a new boundary surfaces classifying algorithm based on the triangular facets collision detection which picks out the splitting surfaces from the boundary set efficiently. An assisted splitting surfaces generation algorithm for separating the curved surfaces from solids was also integrated. After generating a list of splitting surfaces, according to geometrical features and statistics of interfered boundary surfaces of them, a sorting algorithm generates an optimized sequence of splitting surfaces for performing the final decomposition. Furthermore, a new memory management system has been developed that decreases the memory consumption and accelerated the decomposition process as well. A new software architecture makes the decomposition function module more independent and has better compatibility, which could be extended to and integrated with other CAD platforms.

The new decomposition algorithm and the implementation of McCad have been verified with some example components of the European DEMO tokamak. The results show that this advanced McCad version is more efficient and robust and provides more accurate and less complex conversion results. It is thus concluded that the improved McCad new version will be suited for the conversion of complex tokamak models such as the European DEMO or ITER.
On using open source Grasshopper add on for CAD to MCNP conversion

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When simplifying the process of converting CAD models to models suitable for Monte Carlo simulations different approaches have been made. Software for “push of a button” conversion of CAD models to MCNP such as McCad¹⁶, GEOMIT-1¹⁷ and MCAM¹⁸ have been developed. MCAM is tested and used for the ITER experimental fusion reactor CAD conversion. However, prior to direct conversion a simplification of the CAD model is needed and is commonly performed with CAD software such as SpaceClaim. Such approach requires significant amount of time and an expert physicist who can evaluation the degrees of simplification that will not impede final results of the Monte Carlo neutron transport calculations.

Our approach uses a visual scripting language free-ware add-on called Grasshopper (GH) for Rhinoceros 3d (Rhino)¹⁹, a commercial 3D computer graphics application software, to disassemble boundary represented (BREP) CAD structures in to points. The conversion from BREP’s to a collection of points is achieved with different pre-written and newly written plug-ins for GH. Newly created visual scripts than combine three or more points, which are selected by clicking on them by the user from the deconstructed CAD model in Rhino’s graphical user interface, in to MCNP surface input cards, such as the card for an arbitrary oriented plane (p), cylinder or can (rcc), rectangular solid (box) or truncated cone (trc). The surface cards are displayed in Grasshopper and can be copied directly in-to an MCNP input file. Currently surface cards are then combined in to cell cards in an ordinary way²⁰ by the user.

This approach has proved to be simple, fast, effective and the measurements traceable because the user is directly in control of the whole process from selecting the input data to using the surface cards to define cell cards. Work is being done on automating the process of constructing cells for the MCNP input from the surfaces extracted from the original CAD model.

¹⁸ Zeng et al. “Use of MCAM in creating 3d neutronics model for ITER building.” Fusion Engineering and Design, 87(7-8):
Recent photon physics development in the Serpent 2 Monte Carlo code

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The Serpent 2 Monte Carlo code has been used for various fission reactor applications for several years. Recently, some considerable effort has been devoted to extending the capabilities to new fields, including radiation shielding and fusion neutronics. Potential fusion applications for Serpent include heat deposition, material damage, activation and shut-down dose rate calculations. The code has also been coupled to plasma scenarion simulations in an effort to obtain realistic source terms for neutronics calculations. The capabilities have been tested using the C-Lite geometry model of the ITER reactor, which Serpent can read in STL CAD file format without conversion to CSG.

Since photon producing reactions, both prompt and delayed, form an important part of design and safety analyses for fusion reactors, the neutronics codes used in the modeling must also have the capability to perform coupled simulations. The shut-down dose rate calculations discussed previously utilise the radioactive decay source option in Serpent 2, in which the source term for photon transport simulations is formed by a neutron activation calculation combined with photon emission spectra read from ENDF decay data files. This presentation introduces methodology developed for prompt gamma simulations, i.e a coupled neutron-photon transport mode in which prompt gammas are produced in neutron reactions.

Another topic for recent development is variance reduction based on weight windows on automatically generated importance maps. In Serpent 2 the methodology is currently based on a deterministic adjoint solver utilising the response matrix method on a Cartesian or cylindrical mesh.
Development steps towards realistic plasma neutron source for Serpent

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Expanding the development of the Serpent MC code\textsuperscript{21} for use in fusion applications was started in 2015. Neutron production rates for defining probability distributions for sampling source neutrons are calculated by AFSI-ASCOT simulations. As a demonstration case, activation caused by fusion neutrons in ITER geometry and plasma were reported in work\textsuperscript{22}.

In the improved AFSI Fusion Source Integrator code\textsuperscript{23}, the reactions can be evaluated between thermal, fast-thermal and fast-fast species (NBI or RF heating generated ions) by using arbitrary distribution functions of reactive particles from a coupled simulation with the Monte Carlo orbit-following code ASCOT presented in works\textsuperscript{24,25}. The next step in the development of the neutron source is to take into account a realistic energy spectrum of the produced neutrons.

Including the reactions with the energetic ions produces neutron spectra with an energy range much wider than that of typical fusion reactions, potentially causing unexpected effects in the reactor materials which will be studied in forthcoming Serpent calculations.

\textsuperscript{22} P. Sirén, J. Leppänen: Expanding the use of Serpent 2 to fusion applications: development of a plasma neutron source, 2016, PHYSOR conference, Sun Valley, Idaho.
\textsuperscript{23} P. Sirén, J. Varje et al. AFSI Fusion Source Integrator for tokamak fusion reactivity calculations. Under preparation (will be submitted 2016).
\textsuperscript{24} J. A. Heikkinen et al. 2001 Journal of Computational Physics 173 527-548.
\textsuperscript{25} E. Hirvijoki et al. 2014 Computer Physics Communications 185 1310–1321.
The first version of the SUSD3D code was developed in the scope of the EFF project in 1990-ies for the calculations of the sensitivity coefficients and standard deviation in the calculated detector responses or design parameters of interest due to input cross sections and their uncertainties. First-order perturbation theory is used. One-, two- and three-dimensional shielding and criticality problems can be studied. Several types of uncertainties relevant for fusion shielding analysis can be considered, such as those due to neutron and/or gamma multigroup cross sections, including the secondary angular distribution (SAD) and secondary energy distribution (SED) uncertainties. The SAD/SED uncertainties can have important impact for problems where anisotropic scattering (in particular at higher energies) is present, such as in fusion applications.

The SUSD3D code was developed and extensively used in the scope of EC fusion programme since more than 20 years. The latest development and the use within the Fusion for Energy activities will be presented. The extension of SUSD3D, at present using the direct and adjoint fluxes produced by the DOORS, DANTSYS and PARTISN transport codes, to the ATtila codes was studied lately. SUSD3D was used for the sensitivity and uncertainty pre- and post-analysis of several fusion shielding benchmarks performed at the ENEA Frascati Neutron Generator (FNG) such as:

- FNG Bulk Shield benchmark,
- FNG Streaming,
- FNG SiC,
- FNG Tungsten,
- FNG-HCPB tritium breeding modul,
- FNG-HCLL tritium breeding modul;
- FNG Copper benchmark.

The code is available from the OECD/NEA Data Bank and RSICC.
ITER neutron flux monitors benchmark study with NG-24M

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The report describes the results of experimental $^{235}$U and high purity $^{238}$U fission chambers characterization in a "realistic" neutron field and validation of computational models.

Fission chambers with $^{235}$U and extra high purity $^{238}$U (99.9994%) have been designated for ITER total neutron flux monitor. The neutron field was built in TRINITI neutronic laboratory with the 14 MeV neutron generator NG-24M. NG-24M is continuous neutron generator with novel sealed tube. Facility operates at HV range 150-250 kV and up to 1.5 mA ion current producing total neutron yield up to $2 \times 10^{11}$ neutrons per second.

Various kinds of neutron moderators and filters were applied to simulate ITER conditions including loss of coolant and other changes in the diagnostics environment.

The fission chamber features were estimated by criteria of sensitivity, stability and predictability with computation models.

Fission chambers with $^{235}$U converter are extremely sensitive to thermal neutron flux comparing to $^{238}$U, but this causes instability of this detectors in dynamic environment. In particular, one cannot assume that the in-situ calibration will be reliable in long-term period. The computational model, which is necessary for calibration campaign planning and following measurement results interpretation, is questionable for $^{235}$U fission chambers.

The $^{235}$U fission chambers require additional, in some cases unfeasible, efforts to maintain stability and to provide reasonable interpretation of experimental data. At the same time, high purity $^{238}$U fission chambers experimental results satisfy criteria of stability and can be reliably estimated with computational models.

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Neutron measurement instrumentation development at KIT for the EU ITER TBM

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Test Blanket Modules (TBM) will be installed in ITER with the aim to investigate the nuclear performance of different breeding blanket concepts. Currently there is no fully qualified nuclear instrumentation available for the measurement of neutron fluxes and tritium production rates which would be able to withstand the harsh environment conditions in the TBM such as high temperature (up to 650 °C) and, depending on the operation scenario, intense radiation levels.

As partner of the European Consortium on Nuclear Data and Measurement Techniques in the framework of several F4E specific grants and contracts, KIT and ENEA have jointly studied the possibility to develop and test detectors suitable to operate in the EU ITER-TBMs. Here we present an overview of ongoing work on three types of neutron flux monitors under development for the TBMs with focus on the KIT activities.

A neutron activation system (NAS) with pneumatic sample transport could provide absolute neutron flux measurements in selected positions. A test system for investigating activation materials with short half-lives was constructed at the DT neutron generator laboratory of Technical University of Dresden to investigate the neutronics aspects. Several irradiations have been performed with focus on the simultaneous measurement of the extracted activated probes. An engineering assessment of a TBM NAS in the conceptual design phase has been done which considered issues of design requirements and integration.

Within the I_SMART project, funded by KIC InnoEnergy, KIT is developing an online detector based on silicon carbide electronics for the TBMs. The operation of such detectors at TBM relevant temperatures is expected to incur lower accumulated radiation damage to them than at room temperature due to annealing effects. Detectors of several designs have been already irradiated with DT neutrons. Irradiation tests at elevated temperatures have been done and further tests are currently underway.

Self-powered neutron detectors (SPND) are widely applied in fission reactor monitoring, and the commercially available SPNDs are sensitive to thermal neutrons. We are investigating novel materials for SPND which would be sensitive also to the fast neutron flux expected in the TBMs. Only a brief overview of the work is given here, there is a separate presentation on SPND.
Self Powered Detectors for European ITER Test Blanket Modules

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Self-Powered Neutron Detectors (SPND) are electrical devices commonly used in fission reactor cores for neutron-flux mapping. Due to their compactness, ease of maintenance and high-reliability they could serve as a good option for flux monitors in fusion reactors as well. In this work, experimental tests and computational modelling are undertaken to check the possibility of using Self Powered Detectors (SPD) for the measurement of neutron and/or gamma flux in European ITER Test Blanket Modules (TBM). Commercial SPNDs are tailored to the application in thermal fission reactors and are thus, sensitive to low energy neutrons\textsuperscript{28}. New materials are selected based on their neutron activation and physical properties, to replace the central emitter and make the detector responsive to fast neutrons up to D-T neutron energy. A test-device with a flexible design was prepared in-house and experimentally tested with 14 MeV Neutron Generator of Technical University of Dresden (TUD-NG). In the case of considered Beryllium-emitter based SPND, multiple sources of noise, parasitic reactions of high energy neutrons and gammas etc. affected the small current (10^{-13} to 10^{-11} Ampere) output. This device was used to test other emitter-collector combinations as well. A plastic scintillator based counter is under development for measurement of electron emission rates from different material foils including beryllium. In a preliminary test, decay of beta-emitter He-6 from a Be-foil was successfully measured. Alongside experimental, a multi-step Monte Carlo particle transport method for the estimation of the sensitivity of these detectors to the relevant neutron and gamma spectra is utilized to check the appropriate combinations of materials and their sizes.

Global Weight Window Generator Based on Particle Density Uniformity for Monte Carlo Particle Transport Simulation

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Due to the complexity of fusion reactors on geometry and neutron physics, the Monte Carlo (MC) methods have been broadly adopted in fusion nuclear design and analysis. But for calculations that require obtaining a detailed global flux map, such as the shutdown dose rate analysis, analog MC simulations usually cost a prohibitive long run time. To make such analysis computational practicable, it is necessary to adopt an efficient global variance reduction (GVR) method.

This paper proposed a global weight window generator method based on particle density uniformity. For each weight window cell, this method calculates its importance as the expected contribution to the particle density uniformity generated by a unit weight particle entering this cell. This contribution is calculated in a way trying to reach a balance between penetrating deeper by splitting and simulating more source particles per unit time. It also exploits an efficient and fully automatic iteration scheme to speed up the weight window generation. The development of this method is based on the SuperMC code, which is a general, intelligent, accurate and precise simulation software system for the nuclear design and safety evaluation.

To validate the performance of the proposed method, series of tests have been performed with the ITER benchmark, ITER A-Lite and the ITER C-Lite model, calculating the neutron flux over a mesh tally covering the entire reactor. All the tests have showed a substantial increase in computing efficiency compared with the analog case. The highest speedup in the MC figure of merit, ~249 times, is achieved with the ITER A-Lite model. These calculations demonstrate the ability of the proposed method to greatly enhance the efficiency of accurate neutronics simulation of complex models.
Recent Progress on Neutronic Analysis for Diagnostics Procured by JADA

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Neutronic analysis is very important to design diagnostics systems of ITER. In this study, neutronic analysis has been performed to optimize the design of ITER diagnostic systems, procured by Japan Domestic Agency (Poloidal Polarimeter system, Edge Thomson Scattering system, IR Thermography system and Divertor Impurity Monitor system).

Those systems have optical systems. Optical mirrors and Shutters for protection of the first mirror of those diagnostic systems are installed in the equatorial (EQ) and/or Upper port plug. Nuclear heating of those components have been evaluated together with mirror boxes, which store those components, through neutronic analysis (Calculation code; MCNP version 5, Nuclear date library; FENDL 2.1, Calculation model; C-lite version 2). Then, these results were used as input conditions for thermal analysis. Absorbed dose rates of optical windows and optical fibers during ITER operation, which read degradation of optical performance, have also evaluated. It was found that calculated absorbed dose rates for ITER life would be much smaller than operation limit (~ 0.1 MGY) of those components.

Contribution of shutdown dose rate (SDDR) in the interspace area in EQ and Upper port due to installation of a diagnostic system is one of the most important design items. In order to evaluate SDDR, direct 1 step method has been utilised. Contribution of SDDR was evaluated by subtracting SDDR without diagnostic system (full shielded) from SDDR with diagnostics (optical paths are opened). The results suggest that the contributions meet the design target (less than 15 $\mu$Sv/h). However, since the background of SDDR (for example, SDDR without diagnostics) is very high (several hundred $\mu$Sv/h), the contribution has a large margin of error. Reduction of such error is one of important future works to optimize the diagnostic systems.
Development of High Intensity D-T fusion NEutron Generator (HINEG)

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Fusion energy becomes essential to solve the energy problem with the increase of energy demands. Although the recent studies of fusion energy have demonstrated the feasibility of fusion power, it commonly realizes that more hard work is needed on neutronics and safety before real application of fusion energy. A high intensity D-T fusion neutron generator is keenly needed for the research and development (R&D) of fusion technology. However the intensity of D-T neutron generators currently on operation around the world is lower than \(10^{13}\) n/s, which is severely restricting the research capability.

The Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences (INEST•FDS Team) has launched the High Intensity D-T Fusion Neutron Generator (HINEG) project to develop an accelerator-based D-T fusion neutron generator with the neutron yield higher than \(10^{14}~10^{15}\) n/s. The R&D of HINEG includes two phases: HINEG-I and HINEG-II. HINEG-I, which is designed to generate both the steady beam and pulsed beam, has been completed and commissioning since the end of 2015 with the D-T fusion neutron yield of up to \(10^{12}\) n/s. HINEG-II aims at a high neutron yield of \(10^{14}~10^{15}\) n/s neutrons via high speed rotating tritium target system and high intensity ion source. HINEG can be used for research of fusion nuclear technology and safety including the validation of neutronics method and software, radiation shielding and protection, mechanism of materials activation and radiation damage as well as neutronics performance of components. Its application can also be extended to nuclear medicine, radiotherapy, neutron imaging and other nuclear technology applications. This contribution will summarize all the latest progress and future plans for the R&D of HINEG.