

Neutronics calculations for the CTS diagnostics system

Klinkby, Esben Bryndt

Published in:
Book of abstracts - IX ITER Neutronics Meeting

Publication date:
2014

Document Version
Publisher's PDF, also known as Version of record

[Link back to DTU Orbit](#)

Citation (APA):
Klinkby, E. B. (2014). Neutronics calculations for the CTS diagnostics system. In Book of abstracts - IX ITER Neutronics Meeting (pp. 44-44). ENEA.

DTU Library

Technical Information Center of Denmark

General rights

Copyright and moral rights for the publications made accessible in the public portal are retained by the authors and/or other copyright owners and it is a condition of accessing publications that users recognise and abide by the legal requirements associated with these rights.

- Users may download and print one copy of any publication from the public portal for the purpose of private study or research.
- You may not further distribute the material or use it for any profit-making activity or commercial gain
- You may freely distribute the URL identifying the publication in the public portal

If you believe that this document breaches copyright please contact us providing details, and we will remove access to the work immediately and investigate your claim.



Villa Aldobrandini, Frascati; ©Claudio Poggi

BOOK OF ABSTRACTS

IX ITER NEUTRONICS MEETING

JUNE, 24-27, 2014
FRASCATI, ROME, ITALY

<http://www.inm2014.enea.it>



IX ITER NEUTRONICS MEETING
Aula Brunelli ENEA CR Frascati, 24-27 June 2014

AGENDA



IX ITER NEUTRONICS MEETING **Aula Brunelli ENEA CR Frascati, 24-27 June 2014**

AGENDA

DAY 1: Tuesday, 24 June 2014

08:00-08:50 Registration

08:50-09:10 Welcome S. Tosti (ENEA), M. Loughlin (IO)

OVERVIEW SESSION Chair M. Loughlin, R. Villari 09:10-13:15

09:10-09:35 M. Loughlin (IO), Status of ITER neutronics

09:35-10:00 Q. Zeng (INEST), Progress in Fusion neutronics in China

10:00-10:25 M. Sawan (UoW), ITER Neutronics Activities at the University of Wisconsin

10:25-10:50 U. Fischer (KIT), Overview of KIT neutronics activities related to ITER

10:50-11:10 Coffee break

11:10-11:35 L. Packer (CCFE), Overview of CCFE neutronics activities

11:35-12:00 R. Villari (ENEA), Overview of ENEA neutronics

12:00-12:25 D. Leichtle (F4E), Nuclear data related activities relevant to ITER

12:25-12:50 J. Sanz (UNED), Overview of UNED fusion neutronics

12:50-13:15 P. Batistoni (ENEA), JET Project on DT operation technology: Neutronics activities

13:15-14:30 Lunch

Chair U. Fischer, M. Angelone 14:30-17:30

14:30-14:55 I. Kodeli (JSI), Use of SINBAD shielding benchmark database for fusion research

14:55-15:20 S. Putanveetil (IPR), Update on neutronics activities at ITER-India

15:20-15:40 Coffee break

15:40-16:30 K. Ochiai (JAEA), Overview of FNS activities

16:30-17:30 Discussion on "Needs for experimental benchmarking"

DAY 2: Wednesday, 25 June 2014

ANALYSIS SESSION Chair L. Packer, Q. Zeng 08:20-12:25

08:20-08:45 A. Turner (CCFE), Shielding Optimisation of the ICRH Antenna for Shutdown Dose Rate

08:45-09:10 Z. Ghani, Radiation levels in the ITER Tokamak Complex during and after Plasma Operation

09:10-09:35 F. Moro (ENEA), Nuclear Analysis of the ITER Cryoport

09:35-10:00 C. Holland (AMEC), Lower Port Neutronics Analysis

10:00-10:25 S. Sato (JAEA), Shutdown dose rate analysis in transportation of Japanese TBM

10:25-10:50 S. Lilley (CCFE), Nuclear analysis for the ITER Neutral beam cell including intervention scenario

10:50-11:10 Coffee break

11:10-11:35 P. Pereslatsev (KIT), Modelling and Shielding Analysis of the Nuclear Beam Injector Ports in ITER

11:35-12:00 H. Liu (KIT), ITER transfer cask and tunnel dose rate analysis

12:00-12:25 R. Juarez (UNED), Study of Neutron and Shutdown Dose Rate Cross-Talk from Lower to Equatorial Ports in ITER

FNG & ANALYSIS SESSION Chair R. Pampin, J. Sanz 12:25-17:30

12:25-12:50 M. Angelone (ENEA), The 14 MeV Frascati Neutron Generator

12:50-13:15 M. Pillon (ENEA), A methodology to describe the FNG neutron source using the source definition cards (SDEF) in MCNP

13:15-14:30 Lunch

14:30-15:20 Visit to Frascati Neutron Generator

15:20-15:40 Coffee break

15:40-16:05 A. Serikov (KIT), Neutronics analyses for ITER Diagnostics: Cable Looms and Port Plugs EPP and UPP)

16:05-16:30 R. Pampin (F4E), Mitigation of ITER Radiation and Maintenance Dose Levels Inside the Cryostat

16:30-17:30 Discussion on “ Strategy to reduce the shutdown dose rate in maintenance area”

DAY 3: Thursday, 26 June 2014

ANALYSIS SESSION Chair M. Sawan, L. Petrizzi 8.20-12:25

08:20-08:45 R. Juarez (UNED), Optimization of a steel/water based shield for European HCLL and HCPB TBMs and analysis of the resulting Shutdown Dose Rates

08:45-09:10 M. Youssef (UCLA), Occupational Radiation Exposure (ORE) Dose Rate for US DCLL TBM Replacement/Maintenance and Improving Estimates by 3-D Calculations

09:10-09:35 J. Klabacha (PPPL), ITER updated Clite model Comparative Analysis Using Atilla and MCNP

09:35-10:00 L. Petrizzi (EC), Calculations using C-lite model in aid of ITER design issues

10:00-10:25 Q. Yang (INEST), Radiation Dose Assessment in ITER Hotcell During Blanket Replacement

10:25-10:50 A. Suarez (IO), Neutronics for Diagnostics in ITER

10:50-11:10 Coffee break

11:10-11:35 E. Nonbøl (TUD, Nutech), Neutronics calculations for the CTS diagnostic system

11:35-12:00 I. Lengar (JSI), Characterization of the neutron field for the JET torus

12:00-12:25 A. Cufar (JSI), Analysis of Plasma Position Determination Using a Set of Neutron Detectors.

CODE DEVELOPMENT & ANALYSIS OF EXPERIMENTS SESSION Chair K. Ochiai, I. Kodeli 12:25-17:10

12:25-12:50 J. C. Sublet (UKAEA), EASY-II: enhanced physics

12:50-13:15 L. Lu (KIT), Current State of CAD to MC Conversion Tool McCad

13:15-14:30 Lunch

14.30-16:00 POSTER SESSION

- F. Ambrosino (ENEA), ENEA CRESCO HPC Environment for ITER neutronics calculations with MCNP
- A. Turner (CCFE), CCFE development activities in mesh-based shut-down dose rate analysis tools
- F. Ogando (UNED), Absorbed doses in superconducting coils and port cell electronics due to activated water
- C.W. Lee (KAERI) Status of neutronics analyses for Korea HCCR TBS
- B. Caiffi (UoG), Neutronic Analysis of TBM Port #16
- F. Mota (CIEMAT), Lower Port rack neutronics evaluations
- T. Porfiri (ENEA), Last achievements in the assessment of ITER occupational radiation exposure
- D. Portnov (ITER RF), Neutronic Analysis of Divertor Neutron Flux Monitor

16:00-16:20 Coffee break

16:20-16:45 D. Flammini (ENEA), Pre-analysis of the Copper Experiment at the FNG

16:45-17:10 K. Ochiai (JAEA), Basic performance test of TRIPOLI-4.9S code through simple model calculation and analysis of FNS iron benchmark experiment

20:00 Social dinner at Cacciani Restaurant

DAY 4: Friday, 27 June 2014

CODE DEVELOPMENTS SESSION Chair M. Youssef, D. Leichtle 8:20-12:30

08:20-08:45 U. Fischer (KIT), Conclusions of the Benchmarking of the FENDL-3 Neutron Cross Section Data Library for Fusion Applications

08:45-09:10 R. Grove (ORNL), Acceleration of Shutdown Dose Rate Calculations and Uncertainty Propagation Using the Multi-Step CADIS Method

09:10-09:35 I. Kodeli (JSI), SUS3D sensitivity and uncertainty code development and application to fusion

09:35-10:00 S. Yu (INEST), Neutronics Modelling of Blankets and Divertor for ITER C-lite based on MCAM

10:00-10:20 Coffee break

10:20-10:45 J. Song (INEST), Benchmarking of CAD-based SuperMC2.1 with ITER Benchmark Model

10:45-11:10 T. He (INEST), 3D Visual Nuclear Analysis of Radiation Map for ITER PF Coil Maintenance

11:10-11:35 A. Davis (UoW), Summary of code & methods development at UW-Madison

11:35-12:00 Ian M. Davis (Transpire Inc.), Attila4MCNP: GUI Driven, CAD Based, Input and Variance Reduction for MCNP6

12:00-12:50 Ian M. Davis (Transpire Inc.), Attila4MCNP: demonstration

12:50-14:05 Lunch

14:05-14:50 M. Loughlin (IO), Meeting summary, next meeting and closure

14:50-15:10 Coffee break

15:15 Adjourn



IX ITER NEUTRONICS MEETING
Aula Brunelli ENEA CR Frascati, 24-27 June 2014

ABSTRACTS

DAY 1
24 June 2014

Status of ITER Neutronics

M.J.Loughlin¹

¹ *ITER Organization, PSE, Project Engineering and Integration Division
Route de Vinon-sur-Verdon - CS 90 046 - 13067 St Paul Lez Durance Cedex – France*

Email corresponding author: michael.loughlin@iter.org

As the design of ITER is finalised the demands for neutronics analysis shows no signs of slackening. This is because of the several and varied constraints on components design and the need to optimise shielding as the configuration of systems is refined and clarified. This paper will give an overview of the current status of ITER and will provide an up to date summary of the main neutronics issues.

The areas of neutronics can be conveniently divided into the tokamak systems, the tokamak building and the rad-waste and hot-cell buildings. The blanket design is one of the major areas for analysis since this affects the vacuum vessel heating, the TF coil heating and the streaming around the diagnostics ports.

The designs of diagnostics systems and heating systems are proceeding and their status will be reviewed briefly but in this paper the major concern will be how these and others systems integrate. Problems associated with the integration and the long term strategy to address them will be discussed.

The designs of the tokamak, hot-cell and rad-waste building are also being finalised. The status of neutronics analysis will be discussed. The main areas are the shielding for the water cooling systems, the impact on electronics, radiation zoning, port cells and cask transport.

A summary of the report from the sixteenth SATC meeting will also be given.

Progress in Fusion Neutronics in China

Y. Wu, Q. Zeng, Q. Yang, P. Long, J. Song, T. He, S. Yu, L. Hu,
J. Zou, T. Dang, M. Wang, G. Song, FDS Team

*Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences,
Hefei, Anhui, 230031, China*

Email corresponding author: yican.wu@fds.org.cn

Supported by the projects related to ITER and fusion reactors, fusion neutronics in China has been greatly improved in the latest several years. The studies are focused on the development of neutronics codes, nuclear data library, neutronics experiments and nuclear analysis.

A series of neutronics codes, featured to deal with complicated nuclear systems, have been developed by FDS Team, such as Super Monte Carlo Calculation Program for Nuclear and Radiation Process (SuperMC). SuperMC is a CAD-based Monte Carlo program for integrated simulation of nuclear system, making use of hybrid MC-deterministic method and advanced computer technologies. It is designed to perform transport calculation of various types of particles, depletion and activation calculation, and multi-physics coupling calculation. With the supports of Multi-physics Coupling Analysis Modelling Program MCAM, Virtual Reality-based Simulation System for Nuclear and Radiation Safety (RVIS) and Hybrid Evaluated Nuclear Data Library HENDL, SuperMC2.1, the latest version for neutron, photon and coupled neutron and photon transport calculation with integrated modelling and visualization functions, has been developed and validated through series of benchmarking cases, such as the fusion reactor ITER model.

The research platform of highly intensified neutron source and radiation protection in China is under construction. A high intensity D-T neutron generator (HINEG) is under building with neutron intensity nearly 10^{14} n/s, aiming to verify neutronics methodology, calibrate nuclear data library, and test material.

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 2013, a new task for radiological studies around the cargo lift area in hot cell building during plasma operation and cask transfer was finished, providing recommendations for the thickness of shielding doors in level 1. Besides, FDS Team engaged in the design of fusion reactors for about 20 years and has proposed series of fusion reactors and blankets, such as the fusion-driven sub-critical system FDS-I, the

fusion power reactor FDS-II, the fusion-fission hybrid reactor for spent fuel burning FDS-SFB, the Multi-functional experimental reactor FDS-MFX and the Dual Function Lithium Lead liquid breeder Test blanket module for ITER (DFLL-TBM).

ITER Neutronics Activities at the University of Wisconsin

M. Sawan, T. Bohm, A. Davis, P. Wilson, E. Marriott

*Fusion Technology Institute, The University of Wisconsin
1500 Engineering Dr., Madison, Wisconsin 53706 USA*

Email corresponding author: sawan@engr.wisc.edu

The neutronics group at the University of Wisconsin is carrying out several neutronics activities to support the ITER project. We will summarize these activities and discuss the technical findings. Most of the analysis was performed for the blanket modules where several locations with excessive vacuum vessel heating were identified during the final design review (FDR) last April. Several 2-D calculations were performed to evaluate proposed design changes. In addition, detailed 3-D calculations were performed using DAG-MCNP at several locations of concern. These included the NB region (BM14, BM15), the upper ELM coil region (BM11-BM13), the upper port region (BM09-BM11), the BM03/BM04 waveguide region, and BM18 in front of the triangular support. Helium production was determined at blanket connections where rewelding is needed. Analysis with DAG-MCNP is being performed to compare to ATTILA results using the same CAD model with diagnostic port. We are working on generating a CAD version of C-lite that includes detailed modelling of blanket modules to be consistent with the MCNP version of C-lite. Calculations with the newly released photon data library were performed for ITER relevant benchmark problems to assess the impact of calculated nuclear heating. Results indicate a decrease of 2-4% in photon heating. In addition, we continued the effort on improving the workflow for R2S-ACT that couples DAG-MCNP and ALARA. This includes using unstructured mesh elements and improved variance reduction.

IX ITER Neutronics meeting, Frascati, Rome, Italy 24-27 June 2014

Overview of KIT Neutronics Activities Related to ITER

U. Fischer¹

¹ *Karlsruhe Institute of Technology,
Hermann-von-Helmholtz-Platz 1, D 76344 Eggenstein-Leopoldshafen, Germany*

Email corresponding author: ulrich.fischer@kit.edu

An overview is presented of the activities of the Karlsruhe Institute of Technology (KIT) in the field of fusion neutronics related to ITER. The activities include development works on computational tools and methods which are relevant to ITER neutronics as well as design and performance related analyses performed on specific issues in the frame of ITER and F4E contracts.

Overview of CCFE neutronics activities

L. W. Packer, B. Colling, T. Eade, F. Fox, Z. Ghani, S. Lilley, M. Lis, L. Morgan, J. Naish, A. Turner, S. Zheng

CCFE, Culham Science Centre, Oxfordshire, OX14 3DB, UK

Email corresponding author: lee.packer@ccfe.ac.uk

The Applied Radiation Physics (ARP) group at CCFE has extensive experience in radiation transport and activation tools, associated nuclear data and simulation of complex geometric problems. Our team produces comprehensive nuclear and shielding analyses supporting key aspects of the design, optimisation, engineering and safety aspects of a broad range of systems and facilities, notably in support ITER design and provision of regulatory-relevant data.

Here we present a summary of recent activities conducted at CCFE. This presentation will highlight the group's structure, capabilities and broader collaborative activities, both within the UK and internationally. In addition, we introduce our most recent computational neutronics activities in support of ITER, underlying tools & methods development programme and experimental activities.

This work was funded by the RCUK Energy Programme under grant EP/I501045 and the European Communities under the contract of Association between EURATOM and CCFE.

Overview of ENEA neutronics

R. Villari¹, M. Angelone¹, D. Flammini¹, S. Loreti¹, F. Moro¹, M. Pillon¹, S. Podda¹

¹ *ENEA, Unità Tecnica Fusione, 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 performed in ENEA Frascati.

Nuclear analyses have been carried-out with MCNP to support the design of the Test Blanket module Port Plug (TBM PP). Neutron fluxes, nuclear heating, He production and neutron damage have been calculated in all the TBM PP components. Shutdown dose rate calculations have also been performed with Advanced DIS method for different configurations of the TBM PP system, including the study of crosstalk with upper and lower ports.

Extensive nuclear analyses have also been performed for the Lower Cryo-pump ports. Neutrons and gamma fluxes, nuclear heating, absorbed doses on relevant sub-components have been calculated inside the bio-shield and in the port cell. The shutdown dose rate has been assessed with Advanced DIS method to verify the design limits. The contribution of the activated water to the nuclear loads in port cell area has been studied too.

Several computational activities have been performed in preparation for the future DT campaign (DTE-2) at JET including the estimation of in-vessel components activation and the prediction of shutdown dose rate level at the end of DTE-2 with Advanced DIS.

Further nuclear analyses have been carried-out for DEMO in the frame the European Power Plant Physics & Technology Programme (PPPT) and in support to the validation of FENDL-3.0 library.

The pre-analysis for the Copper benchmark experiment at the Frascati Neutron Generator (FNG) for nuclear data validation and studies for the development of neutron and tritium monitors for TBM have been conducted within the Framework Partnership Agreement between the “Consortium for Nuclear Data Studies/Experiments in Support of TBM Activities” and F4E.

Activities devoted to the conceptual design of the “New Sorgentina” Fusion Source (NSFS) project have been carried-out as well.

Nuclear data related activities relevant to ITER

D. Leichtle¹, U. Fischer², M. Angelone³

¹*Fusion for Energy, Barcelona, Spain*

²*Karlsruhe Institute of Technology, Karlsruhe, Germany*

³*ENEA C.R. Frascati, Rome, Italy*

Email corresponding author: dieter.leichtle@f4e.europa.eu

Reliable and accurate nuclear data for fusion technology applications are being developed and validated in world-wide collaborations. A particular and coordinated effort has been devoted in Europe since more than two decades both on the evaluation, processing and benchmarking of nuclear data as well as on the experimental data base required for the validation of fusion relevant libraries such as JEFF and EAF. The overall objective is to develop and validate the predictive capabilities, including reliable uncertainty margins, of nuclear analysis tools in support of the design, construction, licensing, operation and safety case of fusion devices like ITER, DEMO and IFMIF. Fusion for Energy, the European Domestic Agency for ITER, is funding a Nuclear Data Project which covers both theoretical and numerical (nuclear data evaluation, processing and benchmarking), as well as experimental (measurements and experimental validation) activities. It is implemented by Framework Partnership Agreements (FPA) with two nuclear data consortia, formed by European research institutions.

The presentation will give first a layout of the nuclear data activities and the collaboration links to similar international efforts. The main part will focus on recent results as well as on new activities relevant to ITER. On the theoretical and numerical side this includes new evaluations of neutron induced cross section data (Mn, Cu, Ta, Fe, O, Zr,), benchmark studies on fusion relevant experiments for neutron and gamma transport and neutron activation (Pb, Fe, concrete activation, photon heating), development of sensitivity&uncertainty codes and software tools, and development of European reference activation data. On the experimental side, the status of a validation experiment on Cu will be reported as well as R&D activities on candidate TBM nuclear instrumentation. This part is strongly linked to TBM design development and integration as well as further integral testing in a future DT experiment at JET within the framework of EUROfusion.

Overview of UNED fusion neutronics

J. Sanz¹, F. Ogando¹, R. Juarez¹, P. Sauvan¹, J.P. Catalan¹, L. Perez¹, R. Garcia¹, A. Lopez¹

¹ *Departamento de Ingeniería Energética, UNED Calle Juan del Rosal 12, Madrid, Spain*

Email corresponding author: jsanz@ind.uned.es

UNED ITER related activities consist of: i) computational developments and ii) analysis.

R2S-UNED system to compute Shutdown Dose Rates (SDDR) has been optimized for massive calculations. ACAB code has been integrated into a massively parallel execution manager, with almost perfect weak scaling up to 256 simultaneous cores. The sampling process identifying cells inside voxels has been parallelized. Decay gamma sources sum of different calculations (both in cartesian and cylindrical arrangements) are now run in a single calculation. As a result, the computational effort has been reduced in a factor of 10 for a complete-reactor SDDR calculation. Additionally, the graphical capabilities of UNED for nuclear analysis have been extended.

UNED's computational capabilities have been applied to different ITER-related projects. Within F4E tasks, the European TBM and shields have been designed and analyzed from the radiological standpoint supporting the Conceptual Design Review. In addition, the radiation cross-talk phenomenon between Lower Port (LP) and Equatorial Port (EP) has been studied in detail in terms of the SDDR at EP Port Interspace. A strong coupling performance of ITER Ports was found.

Within ITER IO activities, several neutronics issues have been addressed for the components of the EQ#12 including diagnostics located in the EPP and the Port Cell region. The diagnostics performance has been characterized by the usual neutronic quantities: neutron and gamma fluxes, absorbed dose and damage. In the Port Cell room different design options for the bioshield plug have analyzed in terms of the SDDR at 1 day after shutdown. Furthermore, the effect of activated water from the Blanket Shield Module and divertor cooling has been assessed. The energetic decay gammas represent an additional nuclear heat load in the superconducting coils. At the same time, they represent a radiation field in B1 and L1 levels port cells potentially harmful for the electronic equipment.

JET Project on DT Operation Technology: Neutronics Activities

P. Batistoni^{1,2} and EFDA JET contributors

¹ *ENEA, Unità Tecnica Fusione, ENEA C. R. Frascati, via E. Fermi 45, 00044 Frascati (Roma), Italy*

² *JET WPJET3 - DT Technology Project Leader, EUROFusion Consortium*

Email corresponding author: paola.batistoni@enea.it

JET is planning a new full DT campaign (DTE2) in 2017. The proposed 14 MeV neutron budget for DTE2 is nearly an order of magnitude higher than any previous DT campaigns (in JET or TFTR). With this proposed budget, the achievable neutron fluence on the first wall of JET will be up to 1020 n/m², comparable to that occurring in ITER at the end of life in the rear part of the port plug, where several diagnostic components are located. This fluence is higher than practically achievable at existing 14 MeV irradiation facility and, also important, it can be obtained in larger volumes thanks to the volume plasma neutron source. At the expected plasma performance, the neutron flux on the first wall achieves levels not achievable at existing 14 MeV irradiation facility, and is comparable to levels expected in ITER between the blanket and the vacuum vessel (> 1017 n/s•m²).

In the frame of the EUROFusion Consortium Program, a Project (JET3 - DT Technology) has been launched to exploit the unique 14 MeV neutron yields produced in DTE2 and the use of tritium to validate codes, models, assumptions, procedures and data currently used in ITER design thus reducing the related uncertainties and the associated risks in the machine operation.

Neutronics experiments with the fusion relevant conditions in JET, with the appropriate neutron source and environmental complexities are planned with the objective to validate the codes used in ITER design to predict quantities such as neutron flux along streaming paths, activation of materials, shutdown dose rates. An accurate calibration of JET neutron detectors (235U and 238U fission chambers and the in-vessel activation system) at 14 MeV neutron energy will be performed using a DT neutron generator deployed inside the vessel by remote handling.

The 14 MeV neutron flux/fluence will be used to irradiate samples of real ITER materials used in the manufacturing of the ITER main in vessel components, such as W, Be, CuCrZr, 316L(N)-IG. The measured neutron induced activities will be used to validate the calculation predictions. Although the integrated dose and displacement damage would be much lower than in ITER, and

certainly not relevant for structural materials, they would be relevant for functional materials used in ITER heating and diagnostic systems. Irradiation experiments could be performed allowing comparison between available data from fission reactor testing and the effect of a real fusion spectrum from JET, as well as validation of modeling predictions.

The Project activities are supported by extensive neutronics and activation analyses needed to design the experiments and to evaluate the operational and safety aspects.

In the presentation, the planned experiments, their preparation and implementation as well as their scientific objectives will be described.

Use of SINBAD shielding benchmark database for fusion research

I. Kodeli¹, A. Yamaji², R. Grove³

¹ *IJS, Jamova 39, Ljubljana, Slovenia*

² *OECD/NEA, Issy-les-Moulineaux, France*

³ *ORNL, Oak Ridge, USA*

Email corresponding author: ivan.kodeli@ijs.si

In order to preserve and make available the information on the performed radiation shielding benchmarks the Organisation for Economic Co-operation and Development's Nuclear Energy Agency Data Bank (OECD/NEADB) and the Radiation Safety Information Computational Center (RSICC) at Oak Ridge National Laboratory (ORNL) started in the early 1990's the Shielding Integral Benchmark Archive and Database (SINBAD¹) project. Currently, the SINBAD database comprises 100 benchmark compilations and evaluations of relevance to:

- reactor shielding, pressure vessel dosimetry (46)
- fusion blanket neutronics (31)
- accelerator shielding (23)

In addition to characterization of the radiation source, description of the shielding set-up, the instrumentation and the relevant detector measurements, most sets in SINBAD also contain the deterministic or Monte Carlo radiation transport computer model used for interpretation of the experiment and, where available, results from uncertainty analysis. The set of primary documents used for the benchmark compilation and evaluation are provided in computer readable form. The data are organised using the Hyper Text Markup Language (HTML) format with links to the text, figures, Portable Document Format (pdf) documents and computer code inputs. The format has proved to be suitable, general enough and above all easy (and low-cost) to maintain.

The database is intended for different types of users, including nuclear data evaluators, computer code developers, nuclear reactor designers and university students. SINBAD is available from the NEA Data Bank and RSICC. Several hundreds of SINBAD packages have been distributed to date.

¹ I. Kodeli, A. Milocco, P. Ortego, E. Sartori, 20 Years of SINBAD (Shielding Integral Benchmark Archive and Database), *Progress in Nuclear Science and Technology*, Volume 4 (2014)

In 2011 the OECD NEA Nuclear Science Committee (NSC) Working Party on Scientific Issues of Reactor Systems (WPRS) Expert Group on Radiation Transport and Shielding (EGRTS) was started with the mandate (among others), to monitor, steer and support continued development of SINBAD, in cooperation with RSICC. A key objective of the group is to identify, evaluate and preserve experimental data on shielding benchmarks. The Expert Group maintains close links with the International Reactor Physics Experiment Evaluation (IRPhE) and International Criticality Safety Benchmark Evaluation (ICSBEP) projects, as well as in co-ordination with the Joint Evaluated Fission and Fusion File (JEFF) and European Fusion File (EFF) NEA Data Bank projects and with the Shielding Aspects of Accelerators, Targets and Irradiation Facilities (SATIF) community. One of the WPRS/EGRTS' first tasks was to review the current status and future development needs of SINBAD. Several benchmarks available in the database, in particular most fusion experiments, were reviewed to check their completeness and suitability for modern applications, such as nuclear data validation and design studies.

The presentation will focus on the SINBAD benchmark experiments and the associated activity relevant for fusion applications.

Update on neutronics activities at ITER-India

P.V.Subhash¹, S. Jakhar², R. Feder³, S. Thomas¹, D. Agarwal², S. Mishra¹, S. Varshney¹

¹ *ITER-India, Institute for Plasma Research, Gandhinagar, India*

² *Institute for Plasma Research, Gandhinagar, India*

³ *Princeton Plasma Physics Laboratory, Princeton, NJ, USA.*

Email corresponding author: subhashpv@iter-india.org

Neutronics analysis for X-Ray diagnostics system at EPP-11 of ITER is being carried out at ITER-India. This is being performed in two parts, using MCNP and ATTILA. The initial results obtained using MCNP are being recalculated using ATTILA. In this presentation the results with MCNP will be presented initially and ongoing ATTILA results will be followed. Some specific technical details of using ATTILA for large problems like ITER neutronics will be discussed. A brief update of other ongoing neutronics activities will be presented. This includes shielding optimization for X-Ray diagnostic system, port integration of UPP-9 and a status of a CAD to MCNP converter being developed at ITER-India

Overview of FNS Activities - outside ITER -

K. Ochiai¹, M. Ohta¹, S. Sato¹, C. Konno¹

¹*Japan Atomic Energy Agency, Tokai-mura, Naka-gun, Ibaraki-ken 319-1195, Japan*

Email corresponding author: ochiai.kentaro@jaea.go.jp

We have performed various neutronics researches; neutronics experiments with the Fusion Neutronics Source (FNS) facility in Japan Atomic Energy Agency, ITER nuclear analyses and other nuclear analyses. Since the last ITER neutronics meeting, the followings have been carried out; 1) Nuclear analyses for Japan Test Blanket Module, 2) Performance test of the latest TRIPOLI code, 3) Tritium recovery online experiment, 4) Integral experiment for titanium nuclear data benchmarking, 5) FENDL-3.0 benchmark test. The topics 1) – 2) will be presented elsewhere in this meeting. The topic 5) will be published to an IAEA report. We focus on the topics 3) – 4) here.

In the topic 3), we irradiated an experimental assembly simulating the blanket and recovered the tritium from the Li_2TiO_3 pebbles put into the assembly. The activities of the recovered tritium gas (HT) and tritiated water (HTO) were separately measured with a liquid scintillation counter. From our experiment and calculation by using MCNP5-1.40 and FENDL-2.1, the measured tritium corresponded to the calculated tritium production within the experimental error. We have also investigated the influences of the temperature and the water moisture density in the sweep gas line on the ratio of HT/HTO.

In the topic 4) a titanium assembly of 455 mm x 455 mm x 405 mm was covered with Li_2O blocks of 51 or 102 mm in thickness in order to reduce background neutrons inside the titanium assembly. Dosimetry reaction rates of the $^{93}\text{Nb}(n,2n)^{92\text{m}}\text{Nb}$, $^{115}\text{In}(n,n')^{115\text{m}}\text{In}$, $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$ reactions, and fission rate ^{235}U were measured. The measured results were compared with the calculated one by using MCNP5-1.40 code with recent nuclear data libraries. The calculated results with ENDF/B-VII.1 agreed with the measured one the best.

DAY 2
25 June 2014

Shielding Optimisation of the ICRH Antenna for Shutdown Dose Rate

A. Turner¹, D. Leichtle², P. Lamalle³, B. Levesy³, L. Meunier², E. Polunovskiy³

¹ CCFE, Culham Science Centre, Abingdon, Oxon, OX14 3DB, UK

² F4E Fusion for Energy, Josep Pla 2, Torres Diagonal Litoral B3, 08019 Barcelona, Spain

³ ITER Organisation, Route de Vinon sur Verdon, 13115 Saint Paul Lez Durance, France

Email corresponding author: andrew.turner@ccfe.ac.uk

The Ion Cyclotron Resonance Heating (ICRH) antenna is designed to couple RF heating and current drive into the ITER plasma, and will reside in equatorial port plugs 13 and 15. Shutdown dose rates (SDR) within the port interspace are required to be less than 100 $\mu\text{Sv/hr}$ after 106 seconds cooling, in regions where maintenance access is required.

Previous analysis has demonstrated the adequacy of the antenna internal shielding; however a significant contribution to the SDR results from neutrons streaming down the gaps between the port frame and vessel extension, which in common with other ITER port plugs leads to increased activation of surrounding structures. The mitigation of this streaming is the main subject of the presented analyses.

An updated MCNP model of the antenna was created to reflect the latest design, which was integrated into a variant of the B-lite ITER reference model with modified equatorial blankets. Steel shielding plates in the port gaps were proposed to attenuate streaming neutrons, and scoping studies were conducted to assess the effectiveness of several configurations. A configuration was then selected (a front dogleg arrangement), and subjected to high resolution 3-D activation analysis using MCR2S for mesh-coupled transport-activation calculations. It was concluded that the selected configuration was able to reduce the SDR from $\sim 500 \mu\text{Sv/h}$ to $220 \mu\text{Sv/h}$ (still in excess of dose requirements). Approximately 30% of this was due to neighbouring ports; in isolation the ICRH port was shown to result in an SDR of $160 \mu\text{Sv/h}$.

Beams of increased dose rate were observed in the port interspace along the lines of sight of the removable vacuum transmission lines. An angular biasing scheme was implemented into the MCR2S decay gamma source routine to resolve this effect, concluding that the beams were typically $300 - 500 \mu\text{Sv/h}$, which may require further design considerations.

This work was funded by F4E under contract F4E-2008-OPE-002-01-07.

Radiation levels in the ITER Tokamak Complex during and after Plasma Operation

Z. Ghani¹, A. Turner¹, S. Mangham¹, J. Naish¹, M. Lis¹, L. Packer¹, M. Loughlin²

¹ EURATOM/CCFE Fusion Association, Culham Science Centre, Abingdon, Oxon, OX14 3DB, UK

² ITER Organisation, Route de Vinon sur Verdon, 13115 Saint Paul Lez Durance, France

Email corresponding author: zamir.ghani@ccfe.ac.uk

Extensive neutronics and 3-D activation simulations were carried out to assess the levels of radiation throughout the ITER tokamak complex, which comprises the tokamak, diagnostics and tritium handling buildings. The simulated radiation sources included D-T fusion neutrons exiting the cryostat and gamma rays arising from the activation of cooling water and activated pipe chases. Resultant biological dose rates, dose rates to silicon and particle fluxes, for both neutrons and gamma rays, have been calculated and are mapped throughout the complex.

Radiation fields during plasma operation due to activated water show photon biological dose rates approaching 3200 Sv/hr in close proximity to the upper cooling pipes, whilst dose rates on the B2 level of the tokamak complex exceed 0.1 μ Sv/hr inside the diagnostics and tritium handling buildings. Neutrons originating from inside the tokamak have been shown to result in dose rates of less than a Sv/hr inside the port cells (during plasma operation). The dose rate from activated steel pipe chases 10⁶ seconds after shutdown were on the order of 1 μ Sv/hr.

Novel simulation techniques were also applied to assess radiation fields in high fidelity throughout the tokamak complex during cask transfer movements of activated divertor and blanket modules. A custom CCFE boxed surface source routine was used to create source particles from the surface of the cask representative of its contents. An additional custom Fortran MCNP source routine was written to read in route data and sample cask positions on the chosen route. Simulations of integrated dose to electronics based on this 'smeared box source' show that for multiple divertor cask transfers (54 cassette transfers in total), the maximum integrated dose to silicon is 84.5 Gy, observed inside the port cell. The dose inside the tritium handling building is shown to be of the order of 1×10^{-7} Gy.

Nuclear Analysis of the ITER Cryoport

F. Moro¹, R. Villari¹, D. Flammini¹, A. Antipenkov², M. Dremel²,
B. Levesy², M. Loughlin², R. Juarez³, L. Perez³

¹*ENEA, Fusion Technical Unit, Nuclear Technologies Laboratory, Via Enrico Fermi 45, 00044
Frascati, Rome, Italy*

²*ITER Organization, Route de Vinon sur Verdon, 13115 Saint Paul-lez-Durance, France*

³*UNED, Energetic Engineering Department, C/Juan del Rosal 12, Madrid, Spain*

Email corresponding author: fabio.moro@enea.it

ITER will be equipped with 6 torus cryopumps (TCP) positioned inside their housings (TCPH), which are integrated into the cryostat walls at B1 level. Their main function is to reduce the pressure inside the vacuum vessel and extract the helium ashes generated by the fusion reactions and unburned deuterium-tritium fuel that will be processed in the tritium plant and fed back to the plasma.

The present study is focused on a complete assessment of nuclear responses in the Cryopump Ports #4 and #12: the neutronic analysis results provide guidelines for the design of the embedded components, in order to ensure their structural integrity, proper operations and to assess the shielding capabilities.

The geometrical features of the Lower Ports have been singularly integrated in B-lite v3, including the Shielding, TCP structure, the TCPH and all the related elements. Furthermore, the obtained models have been extended beyond the Bioshield to the port door, integrating details of the port cells and building.

Transport calculations have been performed with MCNP5 [2] in order to determine the radiation field inside the Lower Ports, up the port cells: 3-D neutrons and gamma maps have been provided in order to evaluate the shielding effectiveness of the TCPHs. Nuclear heating induced by neutron and photons have been estimated on the TCP and TCPH to assess the nuclear loads during plasma operations, and on Silicon and Stainless Steel in the port cell area to estimate the effect on diagnostics and electronic devices. Moreover, the impact on nuclear responses of the gammas emitted through ¹⁶N β decay in the activated water of the divertor cooling channels has been evaluated. Finally, the shutdown dose rate in the maintenance area of the Lower Ports has been assessed with the Advanced D1S [2] method to verify the design limits.

[1] X5 MONTE CARLO Team, “MCNP—a general Monte Carlo N-Particle transport code: version5 user's guide”, LANL report LA-CP-03-0245, (October 2005).

[2] R. Villari, U. Fischer, F. Moro, P. Pereslavytsev, L. Petrizzi, S. Podda, A. Serikov, “Shutdown dose rate assessment with the Advanced D1S method: development, applications and validation”, paper accepted for the 11th ISFNT Conference (2013), in press

Lower Port Neutronics Analysis

C. Holland¹, A. Kurpesheva¹, A. Tchistiakov¹

¹ AMEC, Booths Park, Chelford Road, Knutsford, Cheshire, WA16 8QZ, UK

Email corresponding author: chris.holland@amec.com

This paper presents the scope of work carried out by AMEC on Neutronics Analysis of the ITER remote handling lower port.

The scope of work is split into two parts.

1. Is to determine shutdown biological dose rate in the port interspaces for an improved shielding design. It includes the conversion of CAD model of Lower port internals into an MCNP input deck, using SpaceClaim/MCAM addition of tallies and variance reduction techniques and production MCNP calculations using the D1S methodology.
2. Is to determine the shutdown biological dose rate in the port cell and gallery due specifically to the activation of components inside the port cell. The port cell walls activation is mainly generated by the presence of ²⁴Na isotope in the activated concrete 24 hours after plasma operation. The analysis requires an application of R2S method with multiple FISPACT runs to calculate dose rates in the port cell and galleries.

Shutdown dose rate analysis in transportation of Japanese TBM

S. Sato¹, K. Ochiai¹, C. Konno¹, H. Tanigawa², T. Hirose², M. Enoda²

¹ *Japan Atomic Energy Agency, Tokai-mura, Naka-gun, Ibaraki-ken 319-1195, Japan*

² *Japan Atomic Energy Agency, Naka-shi, Ibaraki-ken 311-0193, Japan*

Email corresponding author: sato.satoshi92@jaea.go.jp

Shutdown dose rates have been calculated for the ITER JA Water-Cooled Ceramic Breeder Test Blanket Module (WCCB-TBM) port using the partial model. This model includes the WCCB-TBM's components, its frame, shield, flange, port extension, bio-shield, and Ancillary Equipment Unit (AEU). Calculations are done by MCNP5.14, activation code ACT-4, and FENDL-2.1. MCNP geometry input data for the TBM is created from CAD data using the CAD/MCNP automatic conversion code GEOMIT, and other geometry input data is created manually. The "Direct 1-Step Monte Carlo" (D1S) method is adopted for the decay gamma-ray dose rate calculation. It was already reported in the last meeting that the effective dose rates behind the flange at 10^6 seconds after shutdown are 50 - 80 $\mu\text{Sv/h}$, which are lower than 100 $\mu\text{Sv/h}$, the upper limit for human hands-on access for workers performing maintenance. In this meeting we present shutdown dose rates calculated in transportation of the TBM. By using D1S with PWT (Photon Weight Card) function, the decay photons are emitted from only TBM and we calculate the shutdown dose rates due to decay photons from the only TBM. Dose rates in front of the TBM in 1 month, 1 year and 1.5 year after irradiation are 300, 100 and 70 Sv/h, respectively, and they are too high. Dose rates behind the TBM in 1 month, 1 year and 1.5 year after irradiation are 5, 1 and 0.6 Sv/h, respectively, and they are also high. The WCCB-TBM and shield will be inserted to Clite model, and the nuclear analyses will be conducted using this model in this year.

Nuclear analysis for the ITER Neutral Beam cell including intervention scenarios

S. Lilley¹, T Eade¹, F Fox¹

¹ *CCFE, Culham Science Centre, Abingdon, Oxon, OX14 3DB, UK*

Email corresponding author: steven.lilley@ccfe.ac.uk

The ITER neutral beams are located outside the bioshield; the 3 heating neutral beams (HNB) and 1 diagnostic neutral beam (DNB) connect to the plasma chamber via large ducts over 1m². These ducts form very large streaming penetrations through the shielding. This creates both a shielding challenge during plasma operations and could potentially result in large shutdown dose rate in the NB Cell due to activation of NBI components. This work concentrates on the analysis performed in the past year, which includes updates to the model geometry to enable analysis of the DNB, improved variance reduction techniques, and improved shielding around the HNBs. The analysis is performed using MCNP5, MCNP6 as well as FISPACT and MCR2S for shutdown dose calculations. The final model is extremely large with a target geometry accuracy of around 1cm and over 35,000 cells and 80,000 surfaces. The shutdown dose rate needs to be low enough to allow man access to specific locations within the NB cell during interventions in the event of equipment failure.

A key part of the recent work is the development of improved methods to calculate the shutdown dose rate to remote handling equipment during intervention scenarios that require parts of the NBIs to be moved or opened. The improved method modifies the decay photon source routine to make use of MCNP's translation tools. This allows a single decay photon source simulation for both components in their original positions and other components in new positions such as opening of the NBI lid.

The results show that the recommended shielding will reduce the shutdown dose rate and that it is possible to allow manned access to some parts of the NB cell.

Modelling and Shielding Analysis of the Neutral Beam Injector Ports in ITER

P. Pereslavytsev¹, U. Fischer¹, M. Loughlin², Lei Lu¹, E. Polunovskiy², S. Vielhaber¹

¹ *Karlsruhe Institute of Technology, Institute for Neutron Physics and Reactor Technology, Hermann-von-Helmholtz-Platz 1, 76344 Eggenstein-Leopoldshafen, Germany*

² *ITER Organization, Route de Vinon sur Verdon, 13115, Saint Paul Lez Durance, France*

Email corresponding author: pavel.pereslavytsev@kit.edu

The ports for the neutral beam injectors (NBI) in ITER represent openings to the plasma chamber which lead to an intense neutron radiation streaming along the NBI ducts. Neighbouring components such as the vacuum vessel (VV), the NBI duct liners and the toroidal field coil (TFC) magnets are most exposed to this radiation. It is thus required to proof that the assumed shielding around the NBI ports is sufficient for the protection of these components from the impinging intense radiation. To this end a suitable neutronic model of the ITER NBI sector need to be prepared and radiation transport calculations need to be performed to assess the radiation loads to the VV and the TFC in the vicinity of the NBI ducts.

In this work, a new MCNP geometry model of the NBI ports was developed starting from the latest engineering CAD models provided by ITER. The model includes 3 heating (HNBI) ports and the diagnostic port (DNBI) and extends up to the bio-shield. The engineering CAD models were simplified on the CATIA platform according to the neutronic requirements and then converted into MCNP geometry making use of KIT's McCad geometry conversion tool. Finally, the new NBI port model was integrated into an available 80° ITER torus sector model.

The nuclear analysis performed on this model provides the following nuclear responses: the neutron flux distribution in all NBI ports, the nuclear heating distribution in the VV and all NBI ducts; the nuclear heating and radiation loads to the TFC magnets; the radiation damage and gas production in the VV; and the distribution of the shutdown dose rate inside the cryostat. The particle transport simulations were performed using the MCNP5-1.6 code and the FENDL-2.1 data library. Results were generated on high resolution mesh grids making use of MCNP's mesh tally feature. KIT's recent mesh based rigorous two step approach (R2Smesh-2.1) was applied for the calculation of the shutdown dose rate distributions. The Paraview software was used to visualize the mesh distributions on the underlying CAD geometry.

ITER transfer cask and tunnel dose rate analysis

H. Liu, U. Fischer, A. Serikov and P. Pereslavytsev

*Karlsruhe Institute of Technology, Institute for Neutron Physics and Reactor Technology,
Hermann-von-Helmholtz-Platz 1, 76344 Eggenstein-Leopoldshafen Karlsruhe, Germany*

Email corresponding author: haibo.liu@kit.edu

This talk will present the results for two ITER tasks on dose rate analysis. The KIT developed R2Smesh interface code system was used in this study. This code system enables the decay gamma source distribution to be transferred from the irradiation site in the reactor to any other external location for the determination of photon flux and dose rate distributions.

The objective for task one is to determine the maximal dose rate on contact with an automated transfer cask when transporting four different ITER modules: 4 blanket first wall modules, 2 test blanket modules (with frame), one divertor cassette or diagnostic port plug. The analysis shows that the diagnostic generic equatorial port plug (GEPP) module will produce the maximal biological dose rate on the cask surface when it is transported in the automated transfer cask. The maximal biological dose rate is 83.6 Sv/hr on the cask surface with GEPP module. The results partly depend on the positions where the modules locate in the cask.

The objective for task two is to evaluate the biological dose rate outside the tunnel connecting the tokamak complex building and the hot cell facility building while a transfer cask transporting ITER in-vessel modules is passing through the tunnel. The results show that the dose rate is about 68.3 μ Sv/hr at the nearest accessible location outside the tunnel when the cask with the diagnostic GEPP module is crossing the tunnel. In order to examine how the dose rate around the tunnel could be reduced, the tokamak complex building side protrusion was extended to the hot cell facility building side to increase the shielding effect. With the extended protrusion, the dose rate at the nearest accessible location outside the tunnel is 9.31 μ Sv/hr.

This work gives estimate for the dose rate around the transfer cask and the tunnel and gives suggestions on the tunnel design.

Study of Neutron and Shutdown Dose Rate Cross-Talk from Lower to Equatorial Ports in ITER

R. Juarez¹, R. Pampin², A. Antipenkov³, B. Levesy³, F. Moro⁴, A. Suarez³, J. Sanz¹

¹ *Departamento de Ingeniería Energética, UNED Calle Juan del Rosal 12, Madrid, Spain*

² *Fusion for Energy, Calle Josep Pla 2, Torres Diagonal Litoral B3, Barcelona, Spain*

³ *ITER Organization, Route de Vinon sur Verdon, Saint Paul Lez Durance, France*

⁴ *ENEA, C.R. Frascati, Via E. Fermi 45, Frascati, Rome, Italy*

Email corresponding author: rjuarez@ind.uned.es

One of the main issues arising due to activation of ITER port systems is that of occupational radiation exposure (ORE) during planned in-situ maintenance activities in the port inter-space (PI) area. ITER ports and port plugs (PP) become activated by neutron irradiation during plasma operation and the so-called “shutdown” dose rate field (SDR) appears as irradiated materials decay. Project design target for routine maintenance at PI exists: SDR below 100 uSv/hr after 12 days. This is particularly challenging for equatorial ports (EP) due, among others, to the background neutron radiation/activation produced from/by neighbouring components (a.k.a. “cross-talk”).

The study reported here is focused on the evaluation of shielding options to reduce neutron and SDR cross-talk from the LP to the EP in the particular case of a torus cryopump (TCP) lower port (CLP) and a generic diagnostics equatorial port (DEP).

Scoping neutron field analyses were performed of the reference situation plus up to 34 shielding variants based on combinations of six shields located in different parts of the LP, TCP, VV, cryostat and equatorial PI. The study on the reference case showed that about 50% of the neutron flux in the equatorial PI gets there through the CLP.

The most effective shielding variants appeared to be those located inside the LP, near the neutron source, achieving up to a 40% reduction in the flux at EP PI; however these are engineeringly challenging and compromise TCP performance. Performance of other, more manageable and neutral options is poorer, typically <15%. Full SDR field computations are currently being performed on a sub-set of the shielding variants.

The design of the shielding options is subject to further analysis to be taken into account on the basis of engineering feasibility, performance neutrality, and effectiveness in reducing the neutron flux at the equatorial PI.

The 14 MeV Frascati Neutron Generator

M. Angelone, M. Pillon, G. Pagano, S. Loreti, A. Pietropaolo

ENEA, Unità Tecnica Fusione, ENEA C. R. Frascati, via E. Fermi 45, 00044 Frascati (Roma)

Email corresponding author: maurizio.angelone@enea.it

The Frascati Neutron Generator (FNG) is a linear accelerator which accelerates a deuterium beam (300 kV and up to 1 mA) onto a tritiated target to produce 14 MeV neutrons via the D+T fusion reaction (Ref. 1). The maximum neutron yield is $1.0 \cdot 10^{11}$ n/s. The neutron emission is almost isotropic due to the low acceleration potential (300 Kev maximum) but is characterized by the typical beam-target energy-angle distribution.

FNG is in operation since 1992 and during its life has been extensively used for performing a number of neutronics benchmark and mock-up experiments which covered almost all the most important neutronics aspects related to ITER. Among the other, here the ITER mock-ups experiments, the HCLL and HCPB TBM mock-ups, the streaming experiment and the dose rate experiment are recalled since were of direct interest to ITER or commissioned by ITER. The Frascati Neutron Generator (FNG) is a linear accelerator which accelerates a deuterium beam (300 kV and up to 1 mA) onto a tritiated target to produce 14 MeV neutrons via the D+T fusion reaction (Ref. 1). The maximum neutron yield is $1.0 \cdot 10^{11}$ n/s. The neutron emission is almost isotropic due to the low acceleration potential (300 Kev maximum) but is characterized by the typical beam-target energy-angle distribution.

Other studies were related to advanced materials (e.g. SiC and W, Ref. 8,9), materials activation, detectors and electronics development and testing including radiation damage and single event studies.

FNG experiments are routinely analyzed by Monte Carlo analysis based on MCNP code. A subroutine was written and well validated in order to simulate the neutron energy-angle emission from the D-T target. This subroutine is linked to MCNP and used as source routine in the MCNP code.

The paper presents the FNG facility and its applications as well as discusses the MCNP routine used for calculating the neutron emission

A methodology to describe the FNG source using the source definition cards (SDEF) in MCNP/MCNPX.

D. Flammini, M. Pillon, R. Villari

*Associazione EURATOM-ENEA sulla Fusione, ENEA C.R. Frascati, via E. Fermi, 45, 00044
Frascati (Rome) Italy*

Email corresponding author: Mario.pillon@enea.it

The 14 MeV D-T neutron generators like the Frascati Neutron Generator (FNG) have been widely used to perform benchmark experiments in support of the neutronics for the ITER project. The neutron flux, produced by accelerator driven neutron generator as a function of the energy and of the emission angle, has been described by an ad hoc subroutine, developed by the neutronic team at ENEA, for MCNP/MCNPX neutron transport code. This subroutine has been carefully validated by many experimental tests and is currently used to describe the 14 MeV (and also at 2.5 MeV) neutrons produced by FNG [1,2] and, in order to be as faithful as possible to the physical reality of the process of interaction of the deuteron current with the tritiated target, starts from the process of slowing down and diffusion of deuterons inside the target. Then, it generates the neutrons emitted with a certain angle and energy weighing the probability of emission with the DT reaction cross-section. However, the use of this subroutine within the code MCNP/MCNPX requires the compilation of the entire source code. For this reason, users are needed to have the MCNP source code that is usually released to the code developers only. Moreover, it is very difficult to export the FNG source to be used with other codes, for example TRIPOLI, TART, VIM.

To overcome this difficulties we have developed a methodology allowing us to generate an angle and energy dependent neutron distribution, that can be used for the description of the source with the native MCNP/MCNPX source definition (SDEF) or that can be simply exported to be used with other codes. In this work neutron generated with both sources are compared and the results of a previous benchmark experiment [3] has been recalculated and compared with this SDEF MCNP source.

[1] Milocco, A., Trkov, A., Pillon, M., NIM B 271, 2012, pp. 6-12

[2] Pillon, M., Angelone, M., Martone, M., Rado, V., FED 28, IC, 1995, pp.683-688

[3] Batistoni, P., Angelone, M., Fischer, U., Klix, A., Kodeli, I., Leichtle, D., Pillon, M., Pohorecki, W., Villari, R., Nuc. Fusion, 52, I8, 2012, Article number 083014

Neutronics analyses for ITER Diagnostics: Cable Looms and Port Plugs (EPP and UPP)

A. Serikov¹, L. Bertalot², U. Fischer¹, A. Suarez², M. Walsh²

¹ *Karlsruhe Institute of Technology (KIT), Institute for Neutron Physics and Reactor Technology,
Hermann-von-Helmholtz-Platz 1, 76344 Eggenstein-Leopoldshafen, Germany*

² *ITER Organization, Route de Vinon sur Verdon, 13115 Saint Paul Lez Durance, France*

Email corresponding author: arkady.serikov@kit.edu

This presentation draws attention of neutronics experts to the important results of the neutronics analyses recently conducted for several ITER systems in support to the ITER Diagnostics Division. The results of parametric study of shielding improvements affecting SDDR inside the interspace of the diagnostics Generic Upper Port Plug (GUPP) are presented. Five shielding options of the GUPP are summarized. The effect of diagnostic apertures on SDDR is estimated inside the UPP#18 including three diagnostics: VUV-spectrometer, VNC and NAS. The reference results of the GEPP are briefly presented, with an emphasis on the forthcoming analyses for the diagnostics EPP#17 where the Core Imaging X-ray Spectrometer (CIXS) is integrated. As we did before for the GEPP, we are planning to optimize the neutron shielding of CIXS inside the local MCNP models with the boundaries around the EPP. Parametric analyses of gaps and geometrical features of the shielding such as labyrinths, plates, stoppers inside and outside the EPP#17 are planning. The shielding optimization is targeted to minimize the SDDR following the ALARA principle and weight restriction imposed on EPP. The final SDDR analyses will be performed by integration of the optimal shielding configuration of the EPP#17 inside the ITER C-lite with an estimation of radiation cross talks.

Other objects of these neutronic analyses are the ITER cable looms and feedthroughs. The cable looms are part of the ITER cabling system used for connection of diagnostics distributed all around the VV and divertor cassettes. The MCNP tallying approach consists in arranging of voxels in form of small rectangular mesh-tally elements all along the contours of 3D complex shaped cable looms in the B-lite model. The distributions of neutron and photon fluxes, group fluences, nuclear heating, helium gas production, neutron damage, and neutron-induced transmutation of the cable looms and feedthroughs materials are analysed and presented.

Mitigation of ITER Radiation and Maintenance Dose Levels Inside the Cryostat

R. Pampin¹, D. Leichtle¹, M.J. Loughlin², E. Polunovskiy², A. Suarez²

¹*Fusion for Energy, Barcelona, Spain*

²*ITER Organisation, Saint Paul Lez Durance, France*

Email corresponding author: raul.pampin@f4e.europa.eu

Concerns have recently been highlighted at the ITER project regarding radiation levels and dose rates during planned and unplanned maintenance inside the cryostat (port inter-spaces, PFCs). The current understanding is that, under the present configuration, expected levels of these parameters could seriously jeopardise ITER operation. A recently created Neutronics Task Force, which intends to be a collaborative effort between IO and DAs, is in charge of addressing the assessment, design and integration of shielding inside the cryostat for reduction of those levels. This includes large effort to develop and study generic solutions applicable, as much as possible, to all ports and generally across the machine.

This presentation will summarise the current understanding and propose for discussion the main lines of a strategy to coordinate the large effort required to achieve this objective. The basic principles, starting points, lines of action and deliverables of this strategy will be outlined. In particular, the following lines of action are envisaged:

- (i) to reduce the neutron streaming through gaps around UP and EP plugs into the cryostat;
- (ii) to reduce neutron streaming through the lower ports and minimise the cross-talk (both neutron and gamma) between the LPs and the EPs and UPs;
- (iii) to moderate and absorb neutrons by using tailored materials in various other areas, thereby reducing neutron leakage to, and activation in, the cryostat and port inter-spaces;
- (iv) to reduce neutron streaming at the top and bottom of the machine through the gaps between TFCs, which at present cause very high activation levels in these regions and make the potentially needed repair of PF coils very challenging;
- (v) to provide additional shields (absorbers, moderators) both for neutrons and decay gammas within the cryostat; permanent as well as temporal shields should be envisaged.

DAY 3
26 June 2014

Optimization of a steel/water based shield for European HCLL and HCPB TBMs and analysis of the resulting Shutdown Dose Rates

R. Juarez¹, L. Perez¹, P. Sauvan¹, J. Sanz¹, D. Panayotov²

¹ *Departamento de Ingeniería Energética, UNED Calle Juan del Rosal 12, Madrid, Spain*

² *Fusion for Energy, Calle Josep Pla 2, Torres Diagonal Litoral B3, Barcelona, Spain*

Email corresponding author: rjuarez@ind.uned.es

The European Test Blanket Modules (TBM) are based in two approaches: Helium Cooled Lithium Lead (HCLL) and Helium Cooled Pebble Bed (HCPB). The TBMs will be attached to the corresponding TBM shields, and at the same time, they will all be inserted into the Port Plug Frame (PPF). These systems will be hosted in ITER Equatorial Port (EP) #16. Shutdown dose rate (SDDR) at EP Port Interspace (PI) is currently the main design driver for TBM rear shields. The ITER project target is to limit the SDDR to 100 uSv/hr after 12 days cooling. This study presents a feasibility and optimization of a TBM shield made of steel and water, and a nuclear analysis of selected TBM shield design limited to the PI consideration.

In the first part of the study, it is presented an optimization process of the TBM shield using scoping calculations and based on neutron flux attenuation factor performance and activation response for different steel/water distributions inside the shield (more than 20 cases). As a result, a shield divided into three shielding bodies separated by two water tanks, hosting helium and Lithium-Lead pipes was reached. Each shielding body presents a steel/water ratio of 70%/30%, 70%/30% and 25%/75% respectively ordered from the plasma source. This solution offered an attenuation factor of $3 \cdot 10^7$. Then, a constructive solution was generated for this shield, considering water and steel layers, plus helium, PbLi and diagnostics pipes.

In the second part, an MCNP model of the PPF, TBMs and TBM shield is developed and inserted into Blite v3 improved for EP analysis. Then, using the R2S-UNED system, a complete characterization of the SDDR at PI after 12 days of cooling time was performed. The reported results will cover i) SDDR maps at PI, ii) identification of main activation sources contributing to the SDDR, and iii) identification of the main path followed by neutrons to generate the key decay sources.

Occupational Radiation Exposure (ORE) Dose Rate for US DCLL TBM Replacement/Maintenance and Improving Estimates by 3-D Calculations

M. Youssef¹, M. Dagher¹

¹ *School of Engineering and Applied Science, University of California- Los Angeles, CA 90095*

Email corresponding author: youssef@fusion.ucla.edu

In this presentation, the Occupational Radiation Exposure (ORE) for the US Dual Coolant Lithium-Lead Test Blanket Module (DCLL TBM) is estimated based on the latest TBM design and the associated safety evaluation given in the US Preliminary Safety Report (PrSR) issued by Idaho National Laboratory¹⁻². Detailed steps and procedures in maintaining the DCLL TBM and in its removal/installations are described along with the associated duration and level of radiation dose rates. The total annual accumulated ORE is then calculated, assuming the frequency of TBM replacement is one time every three years, and compared to the assessment made earlier to a generic TBM by ITER safety group. If the total ORE assessment to all systems in ITER is 500 person-mSv annually with 26.7% margin, the estimated ORE for current ITER systems is ~367 mSv/a, 1% of which is allocated to each TBM (~3.67 mSv/a). Our current assessment to the DCLL TBM is ~4 mSv/a without accounting for the dose rate background from activation of the TBM structure. When this background radiation effect is added it is expected that the estimated DCLL TBM ORE of ~4 mSv/a will go up and will exceed the allowed limit of 3.67 mSv/a. There is an on-going effort to undertake this improvement in the US DCLL ORE assessment by evaluate the shutdown dose rate (SDDR) in the equatorial port interspace area and in the port cell with the DCLL TBM installed in the detailed C-Lite Model of ITER. The Attila-8.0.0, 3-D, CAD-based Discrete Ordinates Code, along with FENDL-2.1 data library are used in this assessment.

-
- 1) B. J. Merrill, L. C. Cadwallader, M. Dagher, C. P. C Wong, A. Ying, M. Youssef, US Dual Coolant Lead Lithium ITER Test Blanket Module Preliminary Safety Report, INL/EXT-10-18169, Revision 1, June 2010.
 - 2) B. Merrill, L. Cadwallader, M. Dagher, "Occupational Radiation Exposure Analysis of US ITER DCLL TBM", INL/EXT-07-13073, August, 2007

ITER updated C-lite model Comparative Analysis Using Attila and MCNP

J. Klabacha¹, R. Fedder¹, A. Davis², S. Mohammed²

¹ PPPL, PO Box 451, Princeton University, NJ, 08543 USA

² UW-Madison, Madison, Wisconsin 53706 USA

Email corresponding author: jklabach@pppl.gov

Correct ITER nuclear analysis is both important and difficult to obtain. This creates a demand for quick iterative solutions that have been verified with secondary software. Currently, two nuclear analysis programs are being used for nuclear analysis on ITER; MCNP and Attila. MCNP is a Monte Carlo N-particle code while Attila is a three dimensions S_N particle transport code. Initially both codes were verified with a simplified benchmark problem, but have yet to be compared on the large scales that are required for the overall ITER analysis. PPPL in conjunction with the University of Wisconsin-Madison are using the two codes to run a large updated ITER 40° C-lite model. The initial model is created at PPPL using SolidWorks. This model is then loaded into Attila, which has the ability to create a tetrahedral mesh that both Attila and MCNP can use. The model-induced error is removed by the use of the same tetrahedral allowing the code specific results to be compared more precisely. With the compared code results, a new method for Clite simplification is considered. A new simplified model is created that only includes the areas of interest to allow for faster calculation times. A function, which is obtained through the Attila/MCNP comparative analysis, is applied to the simplified model allowing full model results even though the model is greatly simplified. This report covers and compares the PPPL Attila Clite analysis with the UW-Madison MCNP Clite analysis along with Clite model simplification methods.

Calculations using C-lite model in aid of ITER design issues

L. Petrizzi¹, F. Moro², S. Podda², R. Villari²

¹*European Commission DG Research & Innovation, G5, Brussels, Belgium*

²*ENEA, Fusion Technical Unit, Nuclear Technologies Laboratory, Via Enrico Fermi 45, 00044 Frascati, Rome, Italy*

Email corresponding author: luigino.petrizzi@ec.europa.eu

ITER design has to face some design problems in which neutronics has a leading role. Neutronic analyses aim at the calculation of the nuclear loads on some specific components or areas, providing suggestions and recommendations about how to deal with some critical issues.

The present work is focused on different aspects. First of all some advanced tools have been exploited to test their reliability: the newest MCNP ITER model C-lite_V2, variance reduction techniques, and the D1S and AD1S methods for shutdown dose rate calculations. Moreover some nuclear responses have been evaluated in some ITER machine components that have a critical impact on the design of the machine: heat load on the Toroidal Field Coils (TFC) and dose rate after shutdown around Poloidal Field Coils (PFC). The outcome of these studies supplies helpful data to the ITER team and provide an independent check of the neutronic tools to the analysts.

Some advices concerning the use of the high performance computing facilities for ITER neutronic calculations are provided as well.

Radiation Dose Assessment in ITER Hotcell During Blanket Replacement

Q. Yang, Q. Zeng, S. Yu, T. He, C. Chen, Y. Wu, FDS Team

*Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences,
Hefei, Anhui, 230031, China*

Email corresponding author: qin.zeng@fds.org.cn

To optimize the shielding in ITER hotcell building and ensure that the dose rate in each room could meet with the criteria of ITER radiological zoning, the radiation dose in the docking area of hotcell building during blanket replacement need to be assessed.

In the study, the resident activity of a typical blanket module was calculated by coupling the neutron transport with activation calculation using VisualBUS. Three possible proposals for blanket replacement, which are, the cask with only four shielding blocks, or with only four first walls, or with four whole blanket structures, were compared. It was found that when the cask carried with four activated first walls, the radiation dose around was the severest. Then using the automatic modelling program MCAM, which is developed by FDS Team, the combined neutronics model of hot cell building and Tokamak building was created. Along the motion path of the cask, several typical positions were selected and detail dose rate distribution in the docking area was calculated. The results were analysed with the Virtual-reality based Simulation System for Nuclear and Radiation Safety named RVIS, the openings and shielding doors in L1 Level were optimized and reasonable thickness was recommended to abide by the ITER radiological zoning. But further nuclear analysis for other levels in the hotcell building will also need to be performed to verify the shielding capability of concrete doors under different radiation scenarios.

Neutronics for Diagnostics in ITER

A. Suárez¹, J. P. Catalán¹, F. Mota¹, A. Serikov¹, M. Walsh¹,

² *ITER Organization, Saint Paul Lez Durance, France*

² *Karlsruhe Institute of Technology (KIT), Eggenstein-Leopoldshafen, Germany*

Email corresponding author: alejandro.suarez@iter.org

Neutronic calculations are of paramount importance for ITER, being a design driver for many of its components, in particular its Ports and most of its numerous diagnostics. This presentation will give an overview of the neutronic activities within the Diagnostics Division in ITER.

Three main areas concentrate our activities:

- **Shielding:** there is a strong need to provide nuclear shielding to areas where human access is foreseen for maintenance, especially Port Cell area and the challenging Port Interspace area. For this purpose several Task Forces have been created as a collaborative effort between IO and DAs. As owner of many Ports, Diagnostics dedicates significant resources to this task. The different solutions studied will be summarized as well as the strategy for further improvement. In addition, the in-vessel diagnostics need to minimize the impact in the primary shielding components (blankets) in order to avoid significant increases in the heating of the Vacuum Vessel and Superconductive Coils. The current status will be reported.
- **Sensors and diagnostic components:** most of the diagnostics have sensitive components that need to be shielded against radiation to survive. Efficient shielding is not always possible so the radiation levels usually are the decisive factor for the selection of the most appropriate component, which will need to be properly tested against comparable radiation levels.
- **Neutron measurements:** a large set of neutron diagnostics will be implemented in order to measure the neutron production and its emissivity profile. Neutronic calculations are being performed to improve the signal to noise ratios of the neutron cameras installed in the Ports and to evaluate the performance of other neutron detectors. Linked to these activities is the recently started Neutron Calibration Task Force, as a joint effort between IO and DAs, which plans the strategy for the neutron calculations and in-vessel calibration needed, in order to achieve a 10% error in the measurement of fusion power.

Neutronics calculations for the CTS diagnostics system

E. Klinkby¹, E. Nonbøl¹, B. Lauritzen¹, R. Santos²

¹ *DTU Nutech, Frederiksborgvej 399, 4000 Roskilde, Denmark*

² *IST, Avenida Rovisco Pais 1, 1049-001 Lisboa, Portugal*

Email corresponding author: erno@dtu.dk

Recently F4E awarded DTU and IST to partner in the design of a Collective Thomson Scattering (CTS) diagnostic for ITER. The CTS diagnostic utilizes probing radiation of ~60 GHz emitted into the plasma and, using a mirror, collects the scattered radiation by an array of receivers. Having a direct and unshielded view to the plasma, the first mirror will be subject to significant radiation and among the first tasks in the CTS design, is to determine whether the mirror will need active cooling. In order to address this question, a simplified MCNP model of the relevant equatorial port plug #12 was developed based on the full C-lite ITER MCNP model. The first steps toward benchmarking the simplified model to the full C-lite model have been completed, and a preliminary design of the CTS mirror was introduced into the model. Based on this, we have done the first calculations of heat-loads across the mirror and we provide results for a number of different possible designs.

Characterization of the neutron field for the JET torus

I. Lengar¹, L. Snoj¹, A. Čufar¹, B. Syme², P. Batistoni², S. Popovichev², S. Conroy³
and JET EFDA Contributors

JET-EFDA, Culham Science Centre, OX14 3DB, Abingdon, United Kingdom

¹*JSI, Reactor physics division, Jožef Stefan Institute, Ljubljana, Slovenia*

²*CCFE, Culham Science Centre, Abingdon, OX14 3DB, UK*

³*Department of Physics and Astronomy, Uppsala University, Box 516, SE-75120 Uppsala, Sweden*

Email corresponding author: igor.lengar@ijs.si

A new Deuterium-Tritium campaign (DTE2) is planned at JET in 2017, with a proposed 14 MeV neutron budget nearly an order of magnitude higher than any previous DT campaigns. At the expected plasma performance, the neutron flux on the first wall achieves levels comparable to those expected in ITER between the blanket and the vacuum vessel ($> 10^{17}$ n/s·m²). The neutron fluence will be up to 10²⁰ n/m². This level of neutron flux/fluence will offer the opportunity to irradiate samples of functional materials used in ITER diagnostics, and of materials used in the manufacturing of the main in-vessel ITER components, to assess the degradation of the physical properties and the neutron induced activities, respectively.

A number of neutronics experiment and measurements on irradiated samples are planned during the DTE2 campaign. Extensive neutron transport calculations are required in support of these experiments which are performed with the MCNP code and a verified JET model.

The purpose of the present work is to characterize the neutron and gamma ray field inside the JET device during DT plasma operations. An analysis of the neutron/gamma ray flux, energy spectrum and dose rate levels is performed at selected irradiation locations, such as the neutron activation irradiation ends, the new long term irradiation stations located inside the vessel and inside a circular horizontal port, where samples would be exposed to the maximum neutron flux or fluence. The neutron flux and spectrum at different irradiation ends are calculated and compared. The study will offer a unique opportunity for comparing results of calculations with measurements, when the latter become available after the DTE2 campaign.

Analysis of Plasma Position Using a Set of Neutron Detectors

A. Čufar¹, I. Lengar¹, B. Syme², S. Popovichev², L. Snoj¹ and JET EFDA Contributors

¹*JSI, Reactor physics division, Jožef Stefan Institute, Ljubljana, Slovenia*

²*CCFE, Culham Science Centre, Abingdon, OX14 3DB, UK*

³*Department of Physics and Astronomy, Uppsala University, Box 516, SE-75120 Uppsala, Sweden*

Email corresponding author: aljaz.cufar@ijs.si

Determination of plasma position inside the tokamak is important from both a control and a safety point of view. As commercial fusion power plants should be as simple as possible, plasma position determination with a small number of neutron detectors could be appropriate.

A simple MCNP model of a JET tokamak used in the analysis will be described and its main strengths and limitations pointed out. A preliminary feasibility study of fusion plasma position determination using a small set of neutron detectors in four positions around the plasma will be described and the suitability of various detectors discussed. As the actual neutron detectors in tokamaks are positioned behind various layers of shielding or first wall material, the detectors in our model were also positioned behind various layers of the simplified model materials, so the effects of different positioning in the reactor wall could be quantified. The accuracy of the plasma position determination and the time necessary for the instability detection will be assessed and feasibility of using such a system for the plasma monitoring in a DEMO will be discussed.

EASY-II: enhanced physics

J.C. Sublet¹, M. Gilbert¹, J. Eastwood², G. Morgan², A. Koning³, D. Rochman³

¹ *UK Atomic Energy Authority, Culham Science Centre, Abingdon OX14 3DB, United Kingdom,*

² *Culham Electromagnetics Ltd, Culham Science Centre, OX14 3DB, United Kingdom,*

³ *Nuclear Research and Consultancy Group, NL-1755 ZG, Petten, The Netherlands.*

Email corresponding author: jean-christophe.sublet@ccfe.ac.uk

EASY-II is designed as a functional replacement for the previous European Activation System, EASY. It has extended nuclear data and new software, FISPACT-II, written in object-style Fortran to provide new capabilities for predictions of activation, transmutation, depletion and source term of materials under irradiation. The latest FISPACT-II code has allowed us to implement many more features in terms of: energy range; incident particles including alphas, gammas, protons, deuterons and neutrons; physics including neutron self-shielding effects, temperature dependence, pathways analysis, sensitivity and uncertainty quantification using covariance data propagated to all responses (e.g. number density, activity, decay power, dose rates, ingestion and inhalation hazards); and visualization techniques including importance diagrams and nuclide mapping of the full inventory. These capabilities cover all fusion technology and material science needs and all aspects of the operation, safety and environmental assessment of fusion devices. In parallel, the maturity of modern general-purpose nuclear data libraries such as TENDL-2013 encompassing thousands of target isotopes and the capabilities of the latest generation of processing codes such as PREPRO-2013, NJOY2012 and CALENDF-2010 have allowed the FISPACT-II code to be fed with more robust, complete and appropriate data forms: cross-sections with covariance, probability tables in the resonance ranges, prompt and decay kerma, dpa, gas and radionuclide production and 24 decay types. Uniquely, important, often short-lived isomeric states are now properly accounted for. The resulting code and data system, EASY-II, includes many new features and enhancements that directly benefit the validation and verification (V&V) processes being deployed in support of the development of fusion. It has now been extensively tested, having benefited from the experience and feedback of the validation and verification activities performed with its predecessor, and is now ready to be fully integrated into multi-physics simulation toolkits.

Current State of CAD to MC Conversion Tool McCad

L. Lu, Y. Qiu, U. Fischer

*Karlsruhe Institute for Technology (KIT), Institute for Neutron Physics and Reactor Technology,
Hermann-von-Helmholtz-Platz 1 76344 Eggenstein-Leopoldshafen, Germany*

Email corresponding author: lei.lu@kit.edu

McCad is a geometry conversion tool developed at KIT to enable the automatic bi-directional conversions of CAD models into the Monte Carlo (MC) geometries utilized for neutronics calculations (CAD to MC) and, reversed (MC to CAD), for visualization purposes. Because the original algorithms and graphic interface of McCad limited its efficiency, stability and ease of use, related improvements aiming at the algorithms of decomposition, void filling and functions of properties editing etc. have been continuously developed.

This work presents the latest improvements and current state of the McCad new version development, including the more efficient decomposition algorithm avoiding a large number of Boolean operations, and new software architecture and modular design make the function modules are able to be integrated with other graphic software, an example with SALOME graphic platform is demonstrated. Furthermore, new conversion function adapt to MCNP6 tetrahedral mesh input is also implemented, and some applications and validations with this new features are introduced. Furthermore, for solving some problems of incompatibility between McCad and CAD engineering model, the extended programs for geometry repairing and simplification based on SpaceClaim are also being developed, according to the requirements of neutronics modelling. These programs will be necessary processes which will help to achieve the complete conversion from CAD engineering models to MC calculation input file in future.

ENEA CRESCO HPC for ITER neutronics calculations with MCNP

F.Ambrosino¹, F. Moro², S. Podda², R. Villari²

¹ *ENEA UTICT-HPC Department for Computer Science Development and ICT, Piazzale Enrico Fermi, 1, 80055 Portici (Napoli)*

² *Euratom-ENEA Association, UTFUS-TECN Department for Nuclear Fusion, Via Enrico Fermi 45 00044 Frascati (Roma)*

Email corresponding author: fiorenzo.ambrosino@enea.it

The CRESCO HPC infrastructure is composed by several clusters geographically distributed over the ENEA GRID. The bigger cluster is located in the ENEA Research Centre of Portici and it is composed of four sections:

Cresco 1: 42 nodes 4 socket Xeon Quad-Core Tigerton E7330 (2.4GHz), 32 GB RAM; total 672 core.

Cresco 2:

256 nodes Xeon Quad-Core Clovertown E5345 (2.33GHz),

56 nodes Xeon Quad-Core Nehalem E5530 (2.53GHz),

28 nodes Xeon Quad-Core Westmere E5620 (2.40 GHz),

16 GB RAM, total of 2720 core.

Cresco 3: 84 nodes 2-socket 12 core AMD Interlagos 2.4 GHz, 64 GB RAM; total 2016 core.

Cresco 4: 304 nodes dual socket 8-core Intel E5-2670 2.6 GHz, 64 GB RAM; total 4864 core.

The use of Monte Carlo techniques for radiation transport calculations has proven to be essential for the ITER design. MCNP code developed in Los Alamos is one of the reference tool for ITER nuclear analyses. The complexity of the ITER machine geometry and the requirements on accuracy in relevant nuclear quantities calculations need a high level of parallelisation of the process to obtain significant results in reasonable time with low statistical errors.

In the present work the parallel computational performances of the MCNP code in the ENEA CRESCO High Performance Computing (HPC) infrastructure for complex design oriented neutronics calculations using ITER B-Lite V3 model and variance reduction techniques are presented.

CCFE development activities in mesh-based shutdown dose rate analysis tools

A. Turner, L.W. Packer, T. Eade, M.R. Gilbert, Z. Ghani, L.W.G. Morgan, J. Naish, J.C. Sublet

CCFE, Culham Science Centre, Abingdon, Oxon, OX14 3DB, UK

Email corresponding author: andrew.turner@ccfe.ac.uk

MCR2S is the CCFE implementation of the rigorous two-step (R2S) method. MCR2S couples neutron transport mesh tally data from MCNP with the activation capabilities of FISPACT to generate 3-D decay gamma sources and associated inventory data for selected decay times, both of which are important quantities for nuclear analysis of future fusion devices such as ITER and DEMO. The MCR2S program has historically coupled MCNP with FISPACT-2007 and the EAF-2007 activation library (ITER approved activation code and library), and is fully MPI parallelised and validated against other R2S codes and a fusion-relevant experimental benchmark.

CCFE has recently embarked on a programme to update the MCR2S code, which has particularly been driven by the development of FISPACT-II. The successor to FISPACT-2007, FISPACT-II adds significant capability in the areas of uncertainty quantification and support for modern, comprehensive nuclear data formats. It supports activation due to additional particle types – neutron, proton, deuteron, alpha particles and photons, via the use of the ENDF-style libraries e.g. TENDL, as well as supporting the traditional EAF libraries. MCR2S has been adapted to interface with the new FISPACT-II code, paving the way for further advanced capabilities such as photon activation and uncertainty propagation.

Additional improvements have been made in the areas of material-under-mesh assignment. Traditionally performed via production and processing of a PTRAC file, modifications have been made to MCNP to stochastically sample the materials under the mesh, which is now performed in parallel and avoids the need for a serial PTRAC calculation and handling of large files.

MCNP6 supports the use of unstructured mesh geometry, and MCR2S now provides a capability to perform unstructured mesh activation for several element types. Neutron flux data is read from an elemental edit output file, and MCR2S can produce a decay gamma source on the unstructured mesh geometry.

This work was funded by the RCUK Energy Programme (grant number EP/I501045).

Absorbed doses in superconducting coils and port cell electronics due to activated water

F. Ogando, A. López, R. García, J.P. Catalán, R. Juárez, P. Sauvan, J. Sanz

¹ *UNED, C/ Juan del Rosal 12; 28040 Madrid (Spain)*

Email corresponding author: fogando@ind.uned.es

Water used to cooling in-vessel components becomes highly activated. The principal source of radiation is N16 produced in O16 (n,p) threshold reactions. N16 emits high energy gamma rays (~6 MeV) with a half-life of 7 seconds. The cooling water is taken out of the vessel and piped to the upper and lower pipe chases. In doing so the water passes close to the superconducting toroidal field (TF) and poloidal field (PF) coils. This results in heating additional to the prompt radiation from plasma, and is here evaluated in form of specific deposition power and global values per coil and coil case.

The activated water from the divertor is taken through the B1 port cells. The installation of some radiation sensitive equipment is planned in these port cells. An evaluation of the dose radiation condition is required to help in the design, positioning and shielding of the equipment.

It is also important to determine the radiation conditions in the equatorial port cells at L1 level. Gamma-rays from the activated water in B1, neutrons and gamma-rays from the plasma, gamma-rays from the activated machine and building structure, and gamma-rays from activated components in the transfer cask parked in the port cell all contribute to the radiation conditions in the port cell during various modes of operation. The port cell is separated from the gallery in the tokamak building by a port cell door. Above the door is a lintel through which pass various services and opening for air flow. The lintel represents a weak point in the radiation shielding and the evaluation of its performance is required to optimize the dose while maintaining the necessary access.

This work includes results from these analyses calculated with MCNP5, and a comparative analysis of the results with photon libraries MCPLIB04 and MCPLIB84 to assess the impact of the updates to the former and widely used library.

Status of neutronics analyses for Korea HCCR TBS

C.W. Lee¹, Y.O. Lee¹, D.W. Lee¹, S.Cho², M.Y. Ahn²

¹ *Korea Atomic Energy Research Institute, 989 Daeduck-daero, Yuseong-gu, Daejeon 305-353, Korea*

² *National Fusion Research Institute, Gwahangno, Yuseong-gu, Daejeon 305-333, Korea*

Email corresponding author: cwl@kaeri.re.kr

Main design parameters for the TBM (body) had been confirmed based on the case studies. At present, detailed calculation model of the HCCR TBM for neutronics analyses was changed slightly for the breeding zone box and purge gas region. So far, the homogenized TBM-shield model was applied considering cooling water of 20%. Recently, detailed modeling is also on going for cooling pipes inside the TBM-shield. According to the shielding analysis, the design of the TBM-shield may be modified.

For the global analysis, we have produced 2 versions of calculation model based on the B-lite ver. 02 and ver.03 respectively. At present we have been performing update of the global model for the HCCR TBS based on the C-lite model.

R2S method was used in the estimation of the shutdown dose rate. Recently, we have installed the modified MCNP code system using D1S method and tested with ITER benchmark problem. After generation of new global model based on the C-lite, we will use the D1S method for the estimation of the shutdown dose rate.

For the computational code system, MCNP5 ver.1.4 and Cinder ver.1.05 was applied in the neutronics analysis for the HCCR TBS. Recently, FISPACT-2010 was added in the activation analysis and the update of MCNP code from ver.5 to ver.6 is under consideration.

Shutdown Dose Rate Calculation for ITER TBM Port #16

B. Caiffi^{1,2}, R. Villari⁴, L. Giancarli³, M. Loughlin³, M. Osipenko², E. Polunovskiy³,
S.Zheng⁵, D. Panayotov⁶

¹ *Genoa University, Physics Department, Via Dodecaneso 33, 16143 Genoa Italy*

² *INFN, Genoa Section, Via Dodecaneso 33, 16143 Genoa Italy*

³ *ITER Organization, Route de Vinon sur Verdon, 13115 Saint Paul Lez Durance, France*

⁴ *ENEA C.R. Frascati, Via E. Fermi 45, I00044 Frascati, Italy*

⁵ *CCFE, Abingdon Oxfordshire OX14 3DB United Kingdom*

⁶ *Fusion for Energy, Josep Pla, 2, 08019 Barcelona*

Email corresponding author: caiffi@ge.infn.it

The evaluation of the dose rate in the ITER equatorial Port Cell accessible by personnel during long-term shutdown is an important task to validate the main features of the shielding design (e.g., geometry, materials and impurities). The ITER Project target is to limit to 100 $\mu\text{S/h}$ 10^6 seconds (~11 days) the dose rate after the shutdown in the Port Interspace region.

In this work, the recent results of the shutdown dose rate in the TBM Port #16 will be presented. The MCNP model is based on Blite v3, integrated with the description of the Port Plug and of the associated Pipe Forest. The MCNP model used implements the Dummy TBMs. The results were compared with those obtained for Port#2³ (also with Dummy TBMs) and they were found to be in agreement.

The MCNP model with the TBMs of Port #16 (HCPB Helium cooled Pebble Bed and HCLL Helum Cooled Lithium Lead) will be also shown. The model of the TBMs is based on that distributed by F4E⁴. Though, some geometry modifications were necessary to fit the latest design of the TBM Frame.

The shutdown dose rate was calculated using the mesh-based R2S method. Neutron fluxes and spectra were calculated on meshes imposed over the geometry in all the relevant areas, using the weight window technique to achieve the required statistical accuracy. The material description of each mesh voxel was obtained by a preliminary run with a uniform volume source and the MCNP card PTRAC. The activation gamma spectra obtained with FISPACT were used to create a grid of voxel-like sources for the final MCNP run. Using this source, an MCNP gamma transport was performed and the Port Interspace shutdown dose rate was calculated.

³ R. Villari, F.Moro, S. Podda “Finale Report on the Neutronic Analysis of the ITER TBM PP” , IDM JR5EZH (2013)

⁴ S.Zheng, “Further Improvement of the TBS nuclear shielding – MCNP model”, (2011)

Lower Port Rack neutronics evaluations

F. Mota¹, A. Suarez², L. Bertalot², A. Ibarra¹

¹ *Laboratorio Nacional de Fusión, CIEMAT, 28040 Madrid, Spain*

² *ITER Organization, Route de Vinon sur Verdon, 13115 Saint Paul Lez Durance, France*

Email corresponding author: fernando.mota@ciemat.es

In the framework of the service contract “neutronics analysis of ITER diagnostics components CONTRACT NUMBER – 4300000919”, neutronics calculations are being performed to design the low port rack (LP-Rack). The LP-Rack will be located in the low port and it will hold several diagnostics like the low vertical neutron camera (LVNC). The main objective of this contract is to develop neutron transport calculation needed to design the LP-Rack and the LVNC.

Currently, the CAD geometrical model of the LP-Rack has been simplified and converted into MCNP geometrical model and have been also integrated successfully into CLITE MCNP geometrical model. The first part of the LP-Rack has been filled up of B₄C as neutron shielding and the structure is made of SS316LN. Once the LP-Rack was integrated into the CLITE the nuclear heating, damage dose and gas production calculations in the low port area, were got. Hence, these calculations are presented in this report.

On the other hand, the progresses in the LVNC neutronics calculations are also shown.

Last achievements in the assessment of ITER occupational radiation exposure

¹T. Porfiri

¹ *ENEA, Unità Tecnica Fusione, ENEA C. R. Frascati, via E. Fermi 45, 00044 Frascati (Roma), Italy*

Email corresponding author: porfiri@frascati.enea.it

The ORE assessment is for ITER one of the main safety concerns since it heavily impacts both on the experimental program and the maintenance policy of the plant. In the past, the strategy to approach the occupational dose was that of fixing a limit for the maximum dose rate in the zones in which the human presence was foreseen (100 microSv/h) and evaluating the probable contribution by the different systems. At that time, the maintenance procedures available and the similarity with the tasks performed in conventional and fission plants were the basis to build the ORE analysis.

Currently, since detailed neutronics' calculations have become available, it is evident that a strong effort to optimize the assessment is required. In fact, the dose rate in the maintenance zones often exceeds the limit for the workers' access and an ALARA process must iteratively be applied.

The solutions preferred to the purpose of reducing the worker dose are the ones which adopt a remote handling procedure; which reduce the time and the frequencies of the maintenance operations by using components that can last longer, optimize the maintenance procedures, use shields and choose suitable low activation materials.

This work deals with an overview of the last ITER ORE assessment, thus evidencing in which direction the worker dose should be improved, that is to say reduced, with the support of the most recent neutronic studies.

The scope is to supply the neutronics groups a feedback to about the consequences of the activation data on one of the main safety issues pursuant to the regulators.

Neutronic Analysis of Divertor Neutron Flux Monitor

D. Portnov¹, Yu. Kashchuk¹, A. Borisov², N. Deryabina², V. Hripunov²,
D. Markovsky², A. Pashkov²,

¹ *Institution "Project Center ITER" (ITER RF DA), Moscow*

² *National Research Centre "Kurchatov Institute", Moscow*

Email corresponding author: dmitri.portnov@gmail.com

Divertor Neutron Flux Monitor (DNFM) is considered as effective ITER fusion power measurement tool. The computational model for the current DNFM design has been developed based on extended 360 degree ITER MCNP model. The model features and results of neutron and gamma fluxes are presented. DNFM construction nuclear heating and activation effects are studied as well. Induced activity decline with time and spatial distributions are provided.

Pre-analysis of the Copper Experiment at the FNG

D. Flammini¹, R. Villari¹, M. Angelone¹, F. Moro¹

¹*ENEA, Unità Tecnica Fusione, ENEA C. R. Frascati, via E. Fermi 45, 00044 Frascati (Roma)*

Email corresponding author: davide.flammini@enea.it

Copper is an important heat sink material for thermonuclear fusion reactors and it is also used for diagnostic, microwave guides and mirrors in ITER. Within the Framework Partnership Agreement between Fusion for Energy and the “Consortium on Nuclear Data Studies/Experiments in Support of TBM Activities” one of the activity is to design and assembly a block of copper for nuclear data validation by comparing the calculated and the experimental nuclear response under 14 MeV neutron irradiation.

Fusion relevant experiments on copper are lacking: only a measurement of the leakage spectrum on a Cu spherical shell is available [1]. In the present experiment, relevant neutronic responses (e.g. reaction rates, neutron flux spectra, nuclear heating, etc.) will be measured using different experimental techniques (e.g. activation of selected set of foils, neutron and gamma spectrometry, unfolding, dosimeters, etc.) to get the nuclear quantities of interest. The Cu benchmark will be performed at the Frascati Neutron Generator (ENEA-Frascati). The preliminary analyses, performed with MCNP5 [2] Monte Carlo code and presented in this work, assessed the design of the experiment. JEFF 3.1.1 for nuclear transport and IRDF2002 for activation foils have been used as nuclear data library.

The pre-analysis has been performed in order to determine the dimension of the copper block and to optimize the experimental set-up for the measurements. The irradiation conditions and the detectors positions have been investigated in order to get measurable quantities with the detectors under analysis. The design has also been optimized for the background reduction due to the back-scattered neutrons from the walls. The use of a polyethylene shield has been investigated for this purpose.

The results of the pre-analysis show that a 60x70x60 cm³ copper block is the best compromise between performance and costs. It has been demonstrated that the presence of polyethylene is troublesome because it perturbs the neutron spectrum more than the background does.

[1] Alberto Milocco, Andrej Trkov, Ivan A. Kodeli, **The OKTAVIAN TOF experiments in SINBAD: Evaluation of the experimental uncertainties**, *Annals of Nuclear Energy*, Volume 37, Issue 4, April 2010, Pages 443-449,

[2] X5 MONTE CARLO Team, “MCNP—a general Monte Carlo N-Particle transport code: version5 user's guide”, LANL report LA-CP-03-0245, (October 2005)

Basic performance test of TRIPOLI-4.9S code through simple model calculation and analysis of FNS iron benchmark experiment

K. Ochiai¹, M. Ohta¹, S. Sato¹, C. Konno¹

¹ *Japan Atomic Energy Agency, Tokai-mura, Naka-gun, Ibaraki-ken 319-1195, Japan*

Email corresponding author: ochiai.kentaro@jaea.go.jp

A three-dimensional Monte Carlo transport code TRIPOLI has been developed by CEA and has been adopted as one of the ITER neutronics codes. We have tested TRIPOLI code because TRIPOLI code is not always used especially in Japan. In the previous performance test of TRIPOLI-4.4, we pointed out some problems. Recently, the new version of TRIPOLI code, TRIPOLI-4.9S has been released. We test the TRIPOLI-4.9S code.

Two types of calculations are performed with the TRIPOLI-4.9S and MCNP5-1.40 codes. One is a simple model calculation which calculates a leakage neutron spectrum from a sphere with 20 MeV neutron source in the center. The other is an analysis of the iron benchmark experiment performed at JAEA/FNS previously. JEFF-3.1.1 and JENDL-4.0 are used in these calculations as nuclear data libraries. The MCNP5-1.40 code uses the ACE format cross section data. On the other hand, the TRIPOLI-4.9S code uses the original nuclear data file, the PENDF file, and the probability table. The nuclear data library for TRIPOLI code, GALILEE, is used in the calculation with JEFF-3.1.1. For JENDL-4.0, we produce the PENDF file with NJOY-99.336.j4 code and the probability file with CALENDF-2005.69 code.

These calculations were also performed in the previous analysis with the TRIPOLI-4.4 code. There was a problem in the TRIPOLI-4.4 calculation with nuclear data which have the energy-angle distribution data represented by using the Kalbach-Mann systematics. For example, there was a discrepancy around 10 MeV in the leakage neutron spectrum from Nb-93 sphere between the calculated results for JENDL-4.0 with the MCNP5-1.40 and TRIPOLI-4.4 codes. This problem is solved in the calculation with TRIPOLI-4.9S code. And the calculated result with TRIPOLI-4.9S is consistent with that with MCNP5-1.40 for the analysis of FNS iron benchmark experiment. However, it is noted that the point detector in TRIPOLI-4.9S still has more restrictions than that in MCNP5-1.40.

DAY 4
27 June 2014

Conclusions of the Benchmarking of the FENDL-3 Neutron Cross Section Data Library for Fusion Applications

U. Fischer¹, M. Angelone², P. Batistoni², T. Bohm³, K. Kondo¹, C. Konno⁴, M. Sawan³, R. Villari², B. Walker³

¹*Association KIT-Euratom, Karlsruhe Institute of Technology,*

Hermann-von-Helmholtz-Platz 1, D 76344 Eggenstein-Leopoldshafen, Germany

²*Associazione ENEA-Euratom, ENEA Fusion Division, Via E. Fermi 27, I-00044 Frascati, Italy*

³*University of Wisconsin-Madison, 1500 Engineering Dr, Madison, WI 53706, USA*

⁴*Japan Atomic Energy Agency, Tokai-mura, Naka-gun, Ibaraki-ken 319-1195, Japan*

Email corresponding author: ulrich.fischer@kit.edu

FENDL-3 benchmark analyses were performed in a joint effort of ENEA (Italy), JAEA (Japan), KIT (Germany), and the University of Wisconsin (USA) with the objective to test and qualify the neutron induced general purpose FENDL-3.0 data library for fusion applications. The results are now documented in report INDC(NDS)-0631 of the IAEA issued in March 2014. The main conclusion is to recommend to ITER to replace FENDL-2.1 as reference data library for neutronics calculation by FENDL-3.0.

Acceleration of Shutdown Dose Rate Calculations and Uncertainty Propagation Using the Multi-Step CADIS Method[□]

A.M. Ibrahim¹, D.E. Peplow, J.L. Peterson,
and R.E. Grove

¹*Oak Ridge National Laboratory, P.O. Box 2008, Oak Ridge, TN 37831*

Email corresponding author: ibrahimam@ornl.gov

Shutdown dose rate (SDDR) assessments are required everywhere inside the ITER cryostat to evaluate the required waiting period after shutdown before personnel access can be allowed and to identify locations for which human accessibility should be prohibited. Determining the effects on SDDR of important factors such as the cross talk between the different ports with high fidelity is only possible through full-scale simulations involving the complex inner details of the tokamak machine. Accurate full-scale Monte Carlo (MC) calculations of SDDR are impractical using traditional methods because they require high fidelity calculation of space- and energy-dependent neutron fluxes everywhere in the structural materials, of resulting activation products, and of resulting gamma fluxes. Moreover, typical MC SDDR calculations do not consider the impact of uncertainties in the neutron calculation on SDDR uncertainty even though these former uncertainties can dominate the SDDR uncertainty. The Multi-Step Consistent Adjoint Driven Importance Sampling (MS-CADIS) hybrid MC/deterministic method has been proposed to speed up the SDDR MC neutron transport calculation using an importance function that represents the neutron importance to the final SDDR. The MS-CADIS adjoint neutron source can also be used to calculate the SDDR uncertainty resulting from uncertainties in the MC neutron calculation and to assess the degree of undersampling in SDDR calculations because of zero-scoring space-energy mesh tally elements. The MS-CADIS method was applied to the ITER benchmark problem. Compared to the standard global MC variance reduction techniques, the increase in the efficiency of the SDDR neutron MC calculation due to the use of the MS-CADIS method was between 20-70%. The MS-CADIS method also increased the fraction of non-zero-scoring mesh tally elements in the space-energy regions of high importance to the final SDDR.

[□] Notice: This manuscript has been authored by UT-Battelle, LLC, under contract DE-AC05-00OR22725 with the US Department of Energy. The US Government and the publisher, by accepting the article for publication, acknowledges that the US Government retains a non-exclusive, paid-up, irrevocable, world-wide license to publish or reproduce the published form of this manuscript, or allow others to do so, for United States Government purposes.

SUSD3D cross-section sensitivity and uncertainty code development and application to fusion

I. Kodeli

Jozef Stefan Institute, Jamova 39, Ljubljana, Slovenia

Email corresponding author: ivan.kodeli@ijs.si

SUSD3D code calculates 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, i.e. those due to:

- neutron/gamma multigroup cross sections,
- energy-dependent response functions,
- secondary angular distribution (SAD) and secondary energy distribution (SED) uncertainties.

At present, the direct and adjoint fluxes produced by the DOORS, DANTSYS and PARTISN codes can be used to calculate the sensitivities. Extensions to the ATTILA codes are underway.

The code was developed and extensively used in the scope of EC fusion programme since early 90ies. It 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 (ongoing).

The code is available from the OECD/NEA Data Bank and RSICC

Neutronics Modeling of Blankets and Divertors for ITER C-lite Based on MCAM

S. Yu, Q. Yang, Q. Gan, W. Wang, D. Wang, M. Cheng, Y. Luo, Q. Zeng, P. Long, L. Hu, Y. Wu, FDS Team

*Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences,
Hefei, Anhui, 230031, China,*

Email corresponding author: liqin.hu@fds.org.cn

ITER is a complex, multi-national fusion project. In order to meet the subtleties and strict requirements of its nuclear analysis, detailed neutronics reference models of the Tokamak machine need to be established. To establish the latest model ITER C-lite, detailed structure of components such as blankets and divertors, are required to be described and updated. However, the blankets and divertors consist of high-order spline surfaces as well as complex structures, which bring great difficulty for neutronics modelling.

MCAM developed by FDS Team, is designed to be a Multi-Physics Coupling Analysis Modelling System. It is an advanced modelling program aiming to solve modelling problems for multi-physics simulation. It is capable to convert models between CAD systems and multiple Monte Carlo codes including SuperMC, MCNP, PHITS, TRIPOLI, FLUKA and Geant4. It can significantly reduce the manpower and enhance reliability for constructing simulation code input files with complex geometry and it has been used by many institutions in more than 40 countries.

MCAM has been widely applied in ITER modelling. And the neutronics models of blankets No.17, No.18 and the divertor cassettes were created for ITER C-lite based on MCAM. In these models, spline surfaces were replaced by analytic surfaces, unnecessary detail structures were removed, and materials were defined based on the latest design. Validation testing showed the correctness of the generated blankets and divertor neutronics models. These models will be incorporated into ITER C-lite model for nuclear analysis.

Benchmarking of CAD-based SuperMC2.1 with ITER Benchmark Model

J. Song, G. Sun, Z. Chen, H. Zheng, P. Long, L. Hu, Y. Wu, FDS Team

*Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences,
Hefei, Anhui, 230031, China*

Email corresponding author: liqin.hu@fds.org.cn

Super Monte Carlo Calculation Program for Nuclear and Radiation Process (SuperMC) is a multi-physics simulation program mainly based on Monte Carlo (MC) method and advanced computer technology. SuperMC2.1, the latest version, is a CAD-based Monte Carlo transport calculation program for neutron, photon and coupled neutron and photon. In order to validate and demonstrate SuperMC2.1's capability of dealing with complex fusion reactor, especially for the intelligence and advances on complicated geometry processing and correctness of neutron and photon transport in energy range corresponding to fusion reactor, the testing case based on ITER benchmark model was presented in this paper.

The significant advantages of SuperMC2.1 on geometry are: 1) SuperMC2.1 generally starts from directly importing CAD mode and converts geometry internally for further particles transport calculation. 2) SuperMC2.1 adopts hierarchical solid geometry description method assisted with surface description method. Thus, it is not needed for SuperMC to describe or convert cavity cell and avoid particles loss due to the precision of computers. 3) Complex spatial distribution of sources can be converted from CAD model and assigned probability distribution in visualized manner.

The calculation of neutron wall loading on the first wall, neutron flux and nuclear heat in divertor cassettes, nuclear heat in inboard toroidal field coils, neutron/photon flux in equatorial port were calculated with SuperMC and compared with the results of MCNP. The results of two codes were coincident in statistics. The intelligence and advances on complicated geometry processing and correctness of neutron and photon transport in energy range corresponding to fusion reactor in SuperMC were demonstrated.

3D Visual Nuclear Analysis of Radiation Map for ITER PF Coil Maintenance

T. He, S. Zhou, S. Zhang, Q. Zeng, Y. Luo, P. Long, L. Hu, Y. Wu, FDS Team

*Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences,
Hefei, Anhui, 230031, China*

Email corresponding author: liqin.hu@fds.org.cn

Nuclear analysis is becoming more and more data-intensive with increasing complexity of problems and requirement on high accuracy. Huge simulation result data makes data postprocessing and analysis more difficult and inefficient, so it is necessary to find efficient and effective approach for huge data postprocessing and analysis.

RVIS, developed by FDS Team, China, is a virtual-reality based simulation system for nuclear and radiation safety. Based on advanced data visualization and computer graphics technologies, RVIS provides following visual nuclear analysis capabilities: 1) directly support of data postprocessing for multiple codes, such as SuperMC, MCNP and TORT. 2) multiple data visualization including contour, 2D section map, 3D iso-surface, and 3D mesh map. 3) data visualization coupled with calculation geometries, such as 2D map coupled with geometry wire-frame, and mapping result data onto geometry surface. 4) visualization of operation process and real-time dose assessment.

Many visual nuclear analysis cases had been performed based on RVIS. For example, RVIS has been applied in the ITER neutronics tasks finished by FDS Team in 2013, including the estimation of occupational radiation dose during ITER PF4 maintenance and radiological studies around the cargo lift area in ITER hot cell building. The results were 3D visually and interactively analysed by RVIS, shown that RVIS provided a useful and effective approach to improve the efficiency of nuclear analysis.

Summary of code & methods development at UW-Madison

A. Davis, P. PH Wilson, & the CNERG Group

1500 Engineering Drive, Madison, Wisconsin, 53704

Email corresponding author: davisa@engr.wisc.edu

The recent progress on widening the number of the Direct Accelerated Geometry (DAG) toolkit will be reported on, namely FluDAG (Fluka & DAG) and DagSolid (Geant4 & DAG). The University of Wisconsin Unified Workflow (UW2) will be reported on, which facilitates metadata transfer between different codes & workflows. The R2S-ACT workflow, the UW's R2S capability has been significantly upgraded and integrated into the Python for Nuclear Engineers (PyNE) toolkit. A number of other tools will be discussed namely atilla2dag, dag2sn, and make_watertight.

Attila4MCNP: GUI Driven, CAD Based, Input and Variance Reduction for MCNP6

I.M. Davis¹, G.A. Failla¹, R.L. Martz²

¹ Transpire Inc, 6659 Kimball Dr, Ste.E-502, Gig Harbor, WA 98335

² Los Alamos National Laboratory, PO Box 1663, MS A143, Los Alamos, NM 87545

Email corresponding author: ian@transpireinc.com

Attila is an easy to use, CAD based, deterministic radiation transport software suite. It contains an efficient 3D deterministic solver for the Boltzmann transport equation for neutral and charged particles using the linear discontinuous finite element method on an unstructured tetrahedral mesh. The computational mesh from CAD solid models and other input is prepared through a graphical user interface (GUI). Additional features have been added to the GUI to include complete CAD based input preparation for MCNP6 using an unstructured tetrahedral mesh. Recent GUI enhancements have been made to allow set up of consistent source and transport biasing using an Attila adjoint solution for MCNP6, where the geometry is defined for both codes on the same underlying unstructured mesh. A single pass through the GUI defines and executes the Attila adjoint calculation and biased MCNP6 UM calculation automatically. Attila solutions can be post-processed through the GUI, and work is currently being done to extend the post-processing capability for Attila and MCNP6 with a new visualization tool.