International Conference
Nuclear Energy for New Europe 2006
Portorož, Slovenia, September 18-21, 2006
http://www.djs.si/port2006

Computational Methods, Tools, and Data for Nuclear Analyses of Fusion Technology Systems

Ulrich Fischer
Association FZK-Euratom, Forschungszentrum Karlsruhe
Hermann-von-Helmholtz-Platz 1, D- 76344 Eggenstein-Leopoldshafen, Germany
ulrich.fischer@irs.fzk.de

ABSTRACT

An overview is presented of the Research & Development work conducted at Forschungszentrum Karlsruhe in co-operation with other associations in the framework of the European Fusion Technology Programme on the development and qualification of computational tools and data for nuclear analyses of Fusion Technology systems. The focus is on the development of advanced methods and tools based on the Monte Carlo technique for particle transport simulations, and the evaluation and qualification of dedicated nuclear data to satisfy the needs of the ITER and the IFMIF projects.

1 INTRODUCTION

The availability of qualified computational tools and nuclear data for the neutron transport simulation and the calculation of relevant nuclear responses is a pre-requisite to enable reliable design calculations for nuclear facilities. A well-qualified nuclear database and validated computational tools are thus required for quality assured neutronics and activation calculations including uncertainty assessments. Accordingly the international efforts in the field of fusion nuclear technology focus on the development and qualification of computational tools and data required for design analyses of the next step fusion devices ITER, the International Thermonuclear Experimental reactor [1] and IFMIF, the International Fusion Material Irradiation Facility [2].

The objective of this paper is to present an overview of the related Research & Development work conducted at Forschungszentrum Karlsruhe in co-operation with other associations in the framework of the European Fusion Technology Programme. The focus is on the development of advanced methods and tools based on the Monte Carlo technique for particle transport simulations, and the evaluation and qualification of dedicated nuclear data to satisfy the needs of the ITER and the IFMIF projects. In the following these methods, tools and data are described, various applications examples and results of validation analyses are given.

2 COMPUTATIONAL METHODS AND TOOLS

The Monte Carlo method is the standard computational technique for particle transport simulations in fusion technology applications. It allows such a flexible and ease geometry representation that any complex fusion device can be modelled in full 3D geometry without the need for real approximations. The nuclear interaction cross-sections can be used in continuous energy representations as given in the nuclear data files. The accuracy of the calculation thus is affected only by the statistical uncertainty of the calculation itself and the uncertainties of the underlying nuclear cross section data. To satisfy specific needs for the design and analysis of fusion devices such as ITER dedicated methods and computational tools are required though.

2.1 Monte Carlo Based 3D Shut-down Dose Rate Calculations

For safe operation and maintenance of nuclear fusion facilities it is important to be able to predict the induced activation and the resulting shutdown dose rates. This requires a suitable system of codes, data and interfaces which is capable of simulating both the neutron induced material activation during operation and the decay gamma radiation transport after shutdown in full three-dimensional geometry. Such a system, called the rigorous 2 - step (R2S) system [3], has been developed for shutdown dose rate analyses of ITER. The R2S system is based on the use of the MCNP [4] transport and the FISPACT [5] inventory codes linked through a suitable coupling scheme for the automated routing of decay gamma source and neutron flux spectrum distributions.

The R2S approach has been validated on the basis of a benchmark experiment on an ITER shield blanket mock-up performed at the Frascati 14 MeV neutron generator (FNG) [6]. In this experiment, a material assembly made of stainless steel and water-equivalent material has been irradiated with a total of 1.95×10^{15} 14 MeV neutrons. The shutdown dose rates have been measured inside a cavity of the assembly at cooling times ranging from ≈1 h to 20 days after irradiation [7]. Calculated and measured shutdown dose rates are compared in Fig. 1a. An overall satisfactory agreement was obtained over the considered range of decay times.

Further benchmarking was considered necessary by utilizing, as much as possible, a real fusion device with a plasma volume source such as the JET tokamak. Benchmark tests on JET were first made by means of comparison calculations with the direct one step (D1S) approach [8]. This method is based on the assumption that a radioactive nuclide generated during irradiation spontaneously emits the associated decay photons. Neutron and decay photon transport then can be treated in one single Monte Carlo calculation run. When calculating the dose rate, correction factors are applied to account for the proper decay rate of a radioactive nuclide. The shut-down dose rates were calculated for different positions inside and outside the vessel and in the torus hall assuming a representative irradiation scenario. The R2S and D1S results of this calculational benchmark showed agreement within ± 25 % [9]. This was considered satisfactory taking into account the very different approaches.

The outcome of the calculational benchmark suggested a more realistic benchmark exercise on JET. The real irradiation history of D-T and D-D campaigns conducted at JET during the years 1997-98 (DTE 1) were used to calculate the shut-down doses at four different locations (positions 1-4, inside the machine, on the torus hall floor , in contact with the upper coil, in contact with the machine structure) and three different irradiation histories (labelled #1, 9 and 15) with different decay times. The two computational procedures gave results that in general agree with the available measurements within a factor 2 to 3 as shown in Figs. 1 b - d. The comparison was constrained, however, by the rather high uncertainties of the available experimental data recorded by the JET Health Physics team as part of the regular monitoring programme under not well defined conditions. It was thus concluded that a dedicated experiment on JET need to be conducted for the benchmarking. This is currently underway.

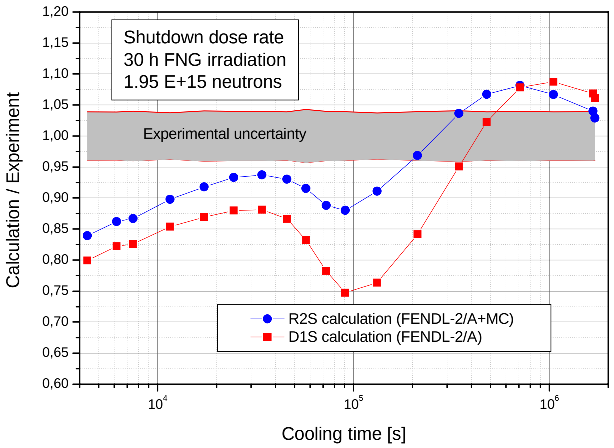


Fig. 1 a: FNG irradiation experiment

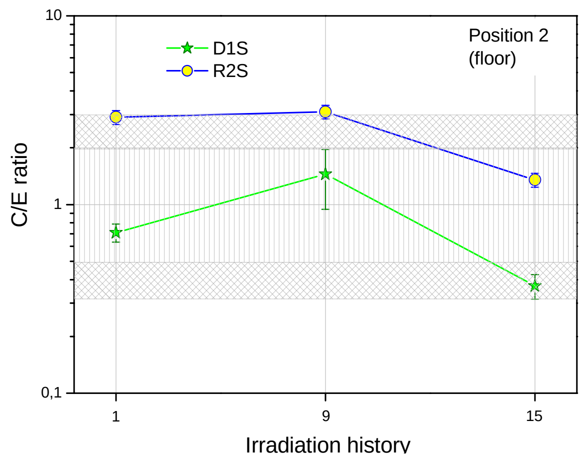


Fig. 1 b: JET dose rate benchmark (pos. 2)

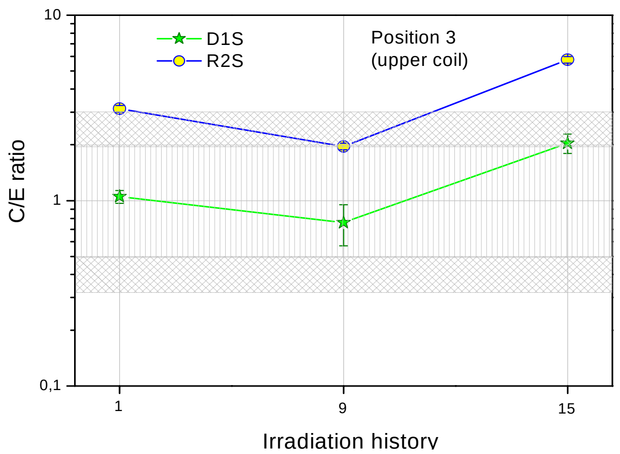


Fig.1c: JET dose rate benchmark (position 3)

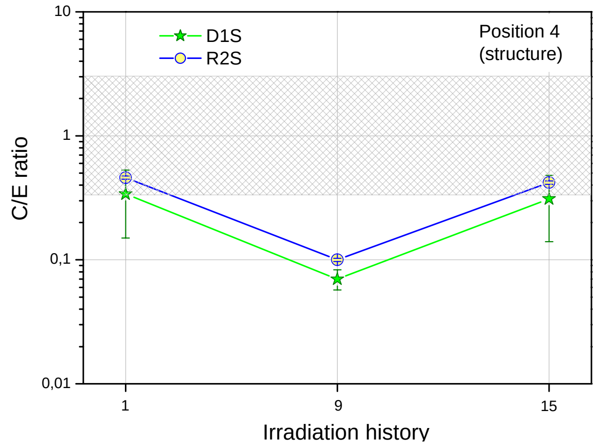


Fig.1d: JET dose rate benchmark (position 4)

Figures 1a-d**:** Comparison of calculated (C) and experimental (E) dose rates obtained for the ITER shut down dose rate experiment at FNG and different positions and irradiation histories at JET.

2.2 CAD Interface for the Monte Carlo Code MCNP

The Monte Carlo technique enables the use of full and detailed 3D geometry models in neutronics calculations. The manual modelling of a complex geometry with a Monte Carlo code, as it is common practice, is an extensive, time-consuming and error-prone task. A more efficient way is to make use of available CAD geometry data in the Monte Carlo calculations. This can be achieved by converting the CAD data into the semi-algebraic representation used by Monte Carlo codes such as MCNP. Suitable conversion algorithms have been previously developed [10] and have been implemented into an interface programme with a graphical user interface (GUI) called McCAD [11].

The McCAD interface programme integrates a CAD kernel, a C++ GUI application framework, and the conversion algorithm. The CAD kernel provides core data structures, algorithms, and data exchange interfaces for neutral CAD data files such as IGES and STEP. The GUI framework provides data structures for visualization and user operations. McCad is also capable of generating a CAD geometry model from an MCNP input deck.

The typical McCAD processing flow is as follows. A geometry model, suitable for neutron transport calculations, is generated with a CAD system. The suitability is determined by the inherent limitations of MCNP with regard to the geometry representation and the adequacy of the model for the neutron transport with regard to complexity. In particular, the

CAD geometry model must be constructed in the boundary representation (B-rep) with algebraic boundary supports such as planes, quadrics and tori, and no free-form surfaces. The only limitation from the algorithmic point of view is that the model should represent a manifold solid, or a collection or an assembly of solids, described in a B-rep data structure. The CAD model is transferred to the interface via data files in the IGES (version 5.3) or STEP neutral format. Both file formats are able to transform B-rep data structures accurately. McCAD then performs automatically model suitability and error checks, and, if possible, repairs. Suitability checks are limited to geometric properties, i.e., the check if the boundary supports are algebraic. Geometrical and topological errors are checked for gaps and overlaps between boundary entities as well as small boundary entities. The next step is then the conversion of the data which is followed by checks for overlaps among solids and their repair. Finally, the model is completed by voids and output in the standard MCNP syntax. This file can be used directly used by MCNP after completion with other required data such as materials specifications, cross section data, source definition, and tally specifications etc.

Successful test applications have been performed for an octant of JET [12] and, more recently, for a 40 degree torus sector of ITER [13] demonstrating the capability for the automated processing of the CAD data and their conversion into a geometry model for Monte Carlo calculations with the MCNP code. Current design applications include the generation and use of MCNP models for the Electron Cyclotron Resonance Heating (ECRH) in the upper port of ITER.

As an example, Fig. 2a shows the CAD model of the 40 degree ITER torus sector consisting of all components relevant for the neutronic analyses. Starting from a CAD geometry model provided by the ITER International Team, Garching, this model was elaborated at FZK with the CATIA V5 software following the guidelines for the generation of CAD neutronics models. The converted MCNP model, generated automatically by McCAD, is shown in Fig. 2b in a vertical cut as provided by the MCNP geometry plotter. It is noted that the conversion process does not introduce any approximations.

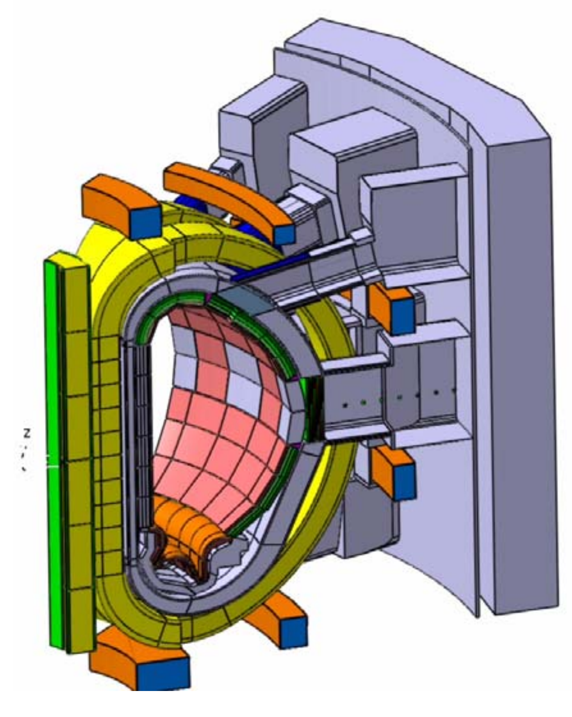


Fig. 2a: CAD model of ITER 40 ° torus sector (CATIA V5)

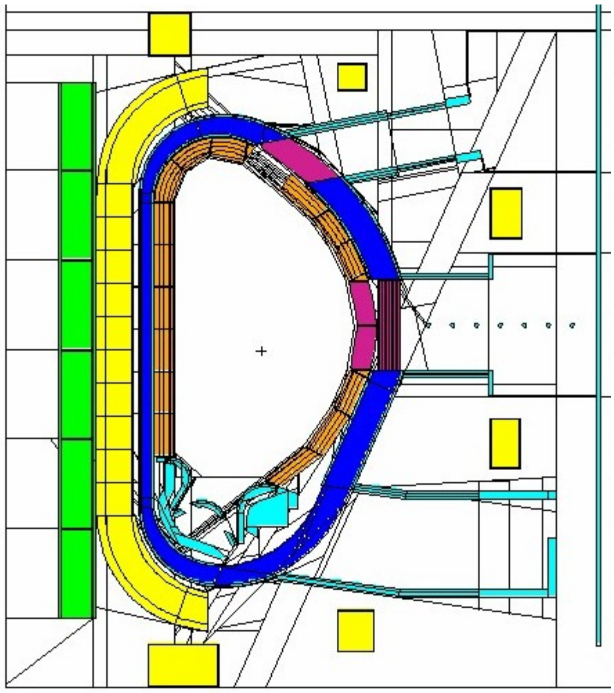


Fig. 2b: Converted MCNP model (2D radial-poloidal cut by MCNP plotter)

The converted geometry was first checked by means of volume comparison calculations for the CAD and the converted models. The volumes of the CAD model could be well reproduced by stochastic MCNP volume calculations using the converted model thus confirming the proper working of the conversion process. A series of transport calculations was next performed starting with the calculation of the neutron wall loading distribution [13]. Further validation calculations including neutron flux and nuclear heating calculations are currently underway in the frame work of a related ITER benchmark exercise.

2.3 D- Li Neutron Source Generation and Transport Simulation

The IFMIF facility will provide an accelerator based D-Li neutron source for high fluence test irradiations of fusion reactor candidate materials. In the IFMIF lithium target, neutrons are generated through the d-Li stripping reaction and various other reaction mechanisms. The neutron source generation must be represented accordingly in the neutron transport calculation. The McDeLicious Monte Carlo code [14] was developed along this guideline to simulate in the transport calculation the neutron generation on the basis of evaluated d + ^{6,7}Li cross sections.

A first set of d + ^{6,7}Li cross section data was evaluated previously in a collaboration of Forschungszentrum Karlsruhe and INPE Obninsk [15]. Tests against thick lithium target experiments showed good agreement for the total and forward neutron yields [16]. Recent measurements of double-differential d+^{6,7}Li cross sections revealed, however, severe deficiencies of the neutron angular distributions and the inability to properly represent the population of residual nucleus excited levels. These results initiated an effort to re-evaluate the d + ^{6,7}Li cross section data applying a new methodology which takes into account compound nucleus reactions, pre-equilibrium processes, stripping and direct interactions [17]. Fig. 4 shows, as an example, the calculated double-differential neutron emission cross section and its breakdown into the different reaction components for 40 MeV incident deuterons.

A series of benchmark calculations was performed with the McDeLicious code to test the new data against experimental thick Lithium target neutron yields. The comparison of measured and calculated forward neutron yields, Fig. 5, shows that McDeLicious with the updated d-Li cross section data is well able to reproduce the experimental results over the entire deuteron energy range from threshold up to 40 MeV. The approaches of McDeLi [19], the predecessor to McDeLicious, and the MCNPX code [20] using the ISABEL intra-nuclear cascade model, give significant worse agreement with the experimental neutron yields.

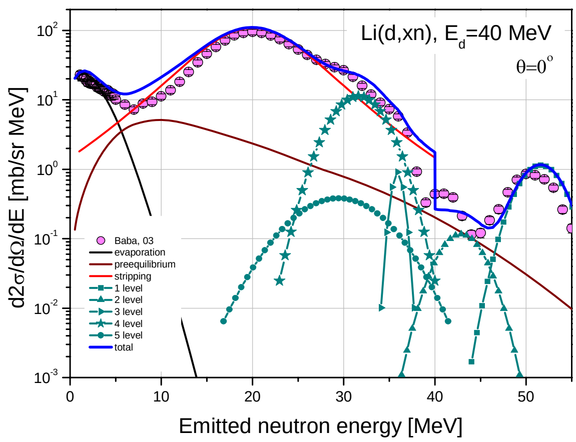


Fig. 4: Measured [18] and evaluated ^{7}Li(d,xn) double differential neutron emission cross section for 40 MeV deuterons.

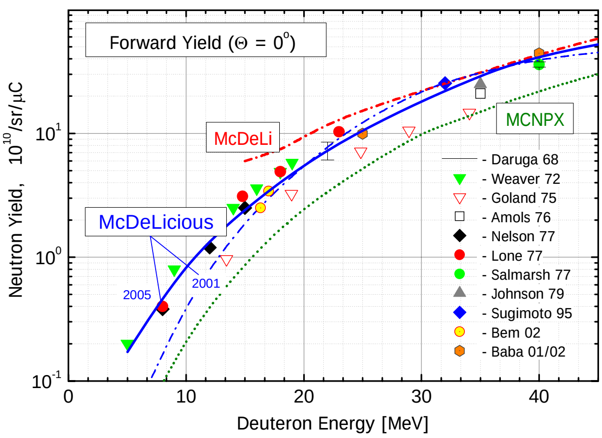


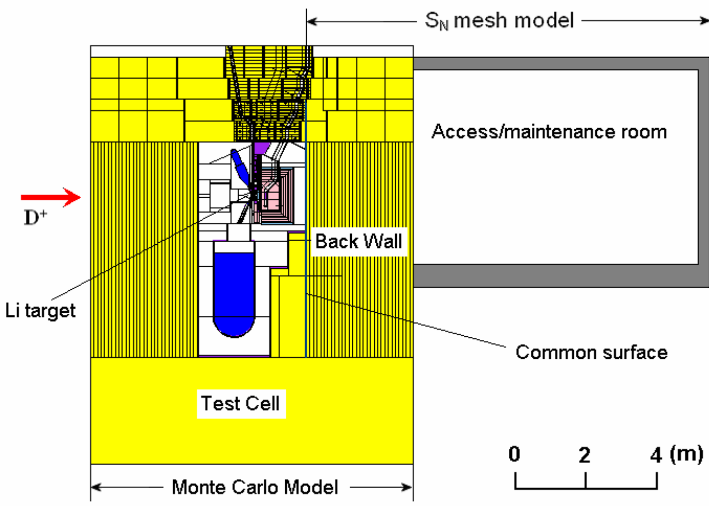
Fig. 5: Measured and calculated thick lithium target forward neutron yields

The comparison of McDeLicious calculations for neutron angular differential yields using the new d-Li evaluation showed that the angular dependence can be satisfactorily predicted over the whole range of measured deuteron energies and secondary neutron angles [17]. The other two approaches show again a worse reproduction of the experimental data set, especially below 20 MeV. The comparison of McDeLicious calculations with measured double-differential thick target neutron yields demonstrates also significant improvements achieved with the updated d-Li evaluation. Thus a clear improvement of the prediction accuracy was obtained for the IFMIF neutron source term simulation with McDeLicious employing the new d + ^{6,7}Li cross section data evaluations.

2.4 Coupled Monte Carlo - Discrete Ordinates Computational Scheme for 3D Shielding Calculations

Shielding calculations of advanced nuclear facilities such as the IFMIF neutron source are complicated due to their complex geometries and their large dimensions, including bulk shields of several meters thickness. To better handle such kinds of shielding problems, a dedicated computational approach for coupled Monte Carlo – deterministic transport calculations has been developed [21, 22]. The Monte Carlo technique is used to simulate the particle generation and transport in and around the neutron source region involving complex geometries while the discrete ordinates method is used to treat the deep penetration problem in the bulk shield.

To enable the coupling of these two different computational methods, a mapping approach has been developed for calculating the discrete ordinates angular flux distribution from the scored data of the Monte Carlo particle tracks crossing a specified surface. The approach has been implemented in an interface programme linking the Monte Carlo code MCNP/McDeLicious and the 3D discrete ordinates code TORT of the DOORS3.2 code package [23].



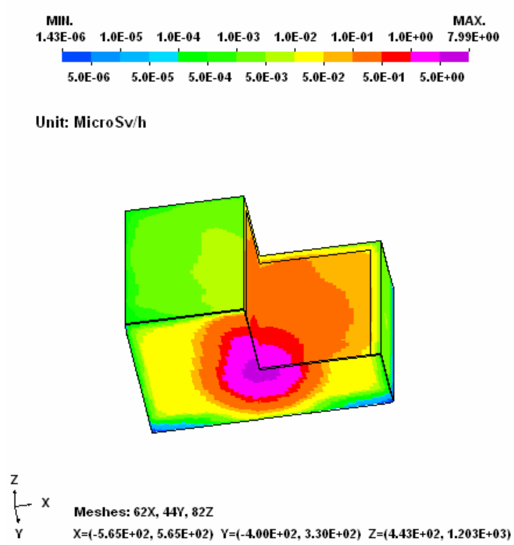


Fig. 5a: Combined MC/S_N geometry model

Fig. 5b: Dose rate distribution in the access/maintenance room

For IFMIF shielding calculations, this approach enables e. g. the use of the detailed geometry model of the test cell in the Monte Carlo calculation along with a proper representation of the D-Li neutron source through the use of the McDeLicious code with the associated d + Li cross-section data. The geometrical model for the coupled MC/S_N

calculation comprises two parts: the test cell with the D-Li neutron source for the Monte Carlo simulation and the maintenance/access room for the S_N calculations (Fig. 5a). The thick concrete wall between the test cell and the maintenance/access room is included in both the Monte Carlo and the S_N mesh model.

The dose rate distribution was assessed across the 3m thick back wall of the test cell and in the maintenance/access room by calculating the neutron and photon flux distributions at IFMIF full power operation. The three-dimensional dose rate distribution obtained for the access/maintenance room is shown in Fig. 5b for the case of a horseshoe-type shield arranged around the test modules. The dose rate is dominated by the neutron radiation and attenuates more than 9 orders of magnitude across the back wall. At the rear of the back wall the dose rate is around 5 μSv/h at maximum which is well below the design limit of 100 μSv/h for work personnel access.

2.5 Monte Carlo Based Sensitivity/Uncertainty Calculations

Sensitivity and uncertainty analysis is a powerful means to assess uncertainties of nuclear responses in neutron transport calculations and track down them to specific nuclides, reaction cross-sections and energy ranges. A method to calculate sensitivities of Monte Carlo point detector responses has been previously developed [24] and implemented in a local version of MCNP4C, called MCSEN. This method has been extended to include sensitivities to secondaries' angular distributions (SAD) [25].

The Monte Carlo based calculation of uncertainties of nuclear responses in the Test Blanket Modules (TBM) of ITER requires the capability to calculate sensitivities for responses by the track length estimator. Suitable algorithms based on the differential operator method were developed to this end and implemented in MCSEN code. This enables the efficient calculation of sensitivities for neutron fluxes and nuclear responses such as reaction rates in a geometry cell of an arbitrary 3D geometry. Sensitivities can be calculated to reaction cross sections, the material density and secondaries' angular distributions. Verification tests were performed by comparing the sensitivities calculated with the track length estimator in a cell and sensitivities calculated by the point detector.

In a first real application, a sensitivity/uncertainty analysis was performed for the neutronics experiment on a mock-up of the HCPB (Helium-Cooled Pebble Bed) TBM employing the track length estimator feature of MCSEN [26]. The dominant tritium production from ^{6}Li, e. g. was shown to be mainly sensitive to the Be cross-sections for the elastic scattering (1.7 – 2.1 %/%) and the (n,2n) reaction (0.7 %/%). The uncertainties of the total tritium production (TPR) due to uncertainties of the reaction cross-sections are at 4% (2σ confidence level) and are dominated by the ^{9}Be uncertainties. The total TPR uncertainties including the data uncertainties, the statistical uncertainties of the Monte Carlo calculation and the experimental uncertainties are in the order of 8 to 10 % (2σ), see Table 1.

Table 1: Total uncertainties (2σ) of expected TPR in the HCPB TBM mock-up

	Stack 1	Stack 3	Stack 5	Stack 7
Calculation				
MC calculations	2.6%	2.8%	3.3%	4.2%
Data uncertainty	3.5%	4.3%	4.0%	3.5%
Calculation + data uncertainty	4.4%	5.2%	5.2%	5.5%
Experiment				
Neutron source uncertainty	6.0%	6.0%	6.0%	6.0%
Measurement uncertainty	3.2%	5.1%	4.8%	5.2%
Total uncertainty (exp. + calc. + data)	**8.1%**	**9.4%**	**9.3%**	**9.6%**

3 NUCLEAR DATA FOR FUSION TECHNOLOGY

The nuclear design of fusion devices relies on the results of neutron and photon transport calculations to provide the neutron/photon flux spectra which form the basis for the calculation of nuclear responses of interest when convoluted with related nuclear data. Appropriate and qualified computational simulations are thus required to ensure that the nuclear responses are reliable. These in turn require appropriate computational tools for the neutron transport simulation along with well qualified nuclear data both for the neutron transport and nuclear responses.

3.1 Nuclear data evaluations and libraries

Various efforts have been conducted on the development of dedicated fusion nuclear data libraries, notably in the European Union (European Fusion File, EFF) and in Japan (JENDL-FF, Japanese Evaluated Nuclear Data Library – Fusion File), see e. g. [27] for an overview. A major international effort was led by the IAEA/NDS when launching the FENDL (Fusion Evaluated Nuclear Data Library) project [28]. The most recent version of the data library, FENDL-2.1 [29], has been adopted as current reference data library for ITER design applications.

In the frame of the EFF project of European Fusion Programme, general purpose data evaluations have been prepared for ^{9}Be, ^{28}Si, $^{46,47,48,49,50}Ti$, ^{nat}V, ^{52}Cr, ^{56}Fe, and $^{58,\ 60}Ni$ including covariance data up to 20 MeV. These EFF evaluations are included in the Joint Evaluated Fission and Fusion File JEFF-3.1 [30]. General purpose data evaluations up to 150 MeV were performed for the stable tungsten isotopes $^{182,183,184,186}W$ [31] and ^{181}Ta [32]. These are currently being complemented by covariance data.

3.2 Validation experiments and analyses

Testing and validation are essential in the process of assuring the quality assurance of the nuclear data evaluations for application calculations. This is achieved through integral benchmark experiments and their computational analyses both for transport and activation experiments.

The current focus of the experimental benchmark effort in the EU is on neutronics TBM mock-ups. The first experiment of this kind has been performed at FNG on a TBM mock-up of the HCPB breeder blanket to check and validate the capability of the neutronic codes and nuclear data to predict its tritium production capability [33]. The mock-up replicates the main characteristics of a breeder insert of the TBM HCPB in ITER. It consists of a stainless steel box with outer dimensions of 31x31x29 cm, filled by alternating layers of breeder material (Li_2CO_3) and neutron multiplier (Be), see Fig. 6 for the MCNP model.

The tritium generated during irradiation in a series of Li_2CO_3 pellets located at different penetration depths in the mock-up was found to be underestimated by the calculations by 5 to 10% on average independent on the nuclear data used, see Figs. 7 a, b for pellet stacks1 & 2 and 7 & 8 located in the front and the back of the assembly, respectively. The obtained results indicate that design calculations for the tritium breeding ratio (TBR) of fusion power reactors employing a HCPB type breeder blanket are conservative. Thus an additional TBR margin is provided which allows compensating for potential other uncertainties.

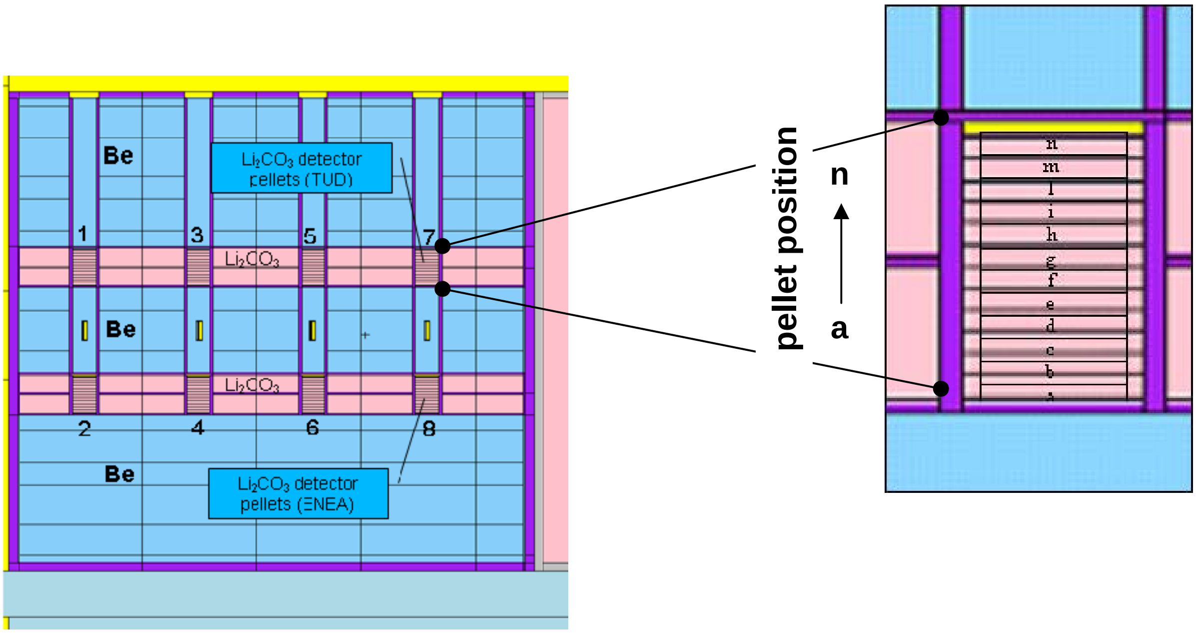


Figure 6: MCNP model of the TBM-HCPB mock-up. Vertical cut through the center of the block, showing the two sets of symmetrical detector pellet stacks.

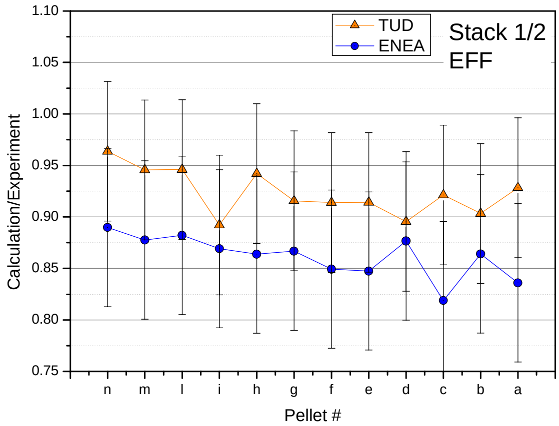


Fig. 7a: Pellet stacks 1 & 2

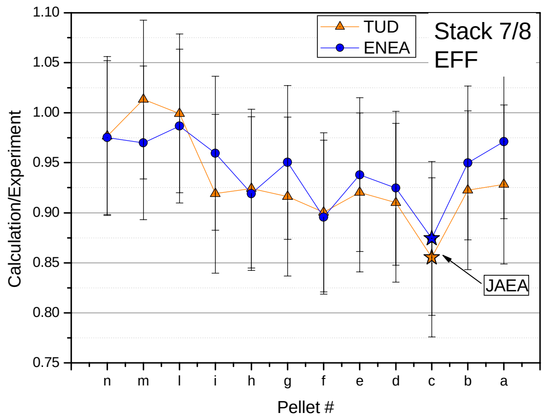


Fig. 7b: Pellet stacks 7 & 8

Figures 7a,b: Ratios of calculated (C) and experimental (E) tritium activities measured in the pellets of stack numbers 1& 2 and 7 & 8.

4 SUMMARY AND CONCLUSIONS

An overview has been presented of the R&D work conducted at Forschungszentrum Karlsruhe in co-operation with other associations in the framework of the European Fusion Technology Programme on the development and qualification of computational tools and data for nuclear analyses of Fusion Technology systems. The focus has been on various advanced methods and tools based on the Monte Carlo technique for particle transport simulations, and the evaluation and qualification of dedicated nuclear data developed to satisfy the needs of the ITER and the IFMIF projects.

ACKNOWLEDGMENTS

This work, supported by the European Communities under the contract of Association between EURATOM and Forschungszentrum Karlsruhe, was carried out within the framework of the European Fusion Development Agreement.

REFERENCES

[1] Y. Shimomura, The present status and future prospects of the ITER project, J. Nucl Mat. 329-333, 5- 11 (2004)

[2] IFMIF International Team, IFMIF Comprehensive Design Report, January 2004

[3] Y. Chen, U. Fischer, Rigorous MCNP based shutdown dose rate calculations: Computational scheme, verification calculations and applications to ITER, Fus. Eng. Des. 63-64 (2002), 107-114

[4] J. F. Briesmeister (ed.), MCNP - A General Monte Carlo N-Particle Transport Code, Version 4C, Los Alamos National Laboratory, Report LA-13709-M, April 2000

[5] R. A. Forrest, J-Ch. Sublet, FISPACT 99: User Manual, UKAEA Fusion, Report UKAEA FUS 407, December 1998

[6] P. Batistoni, M. Angelone, L. Petrizzi, M. Pillon, Benchmark Experiment for the Validation of Shut-Down Activation and Dose Rate in a Fusion Device, J. Nucl. Sci. Techn., Suppl. 2, p. 974-977 (2002)

[7] K. Seidel, Y. Chen, U. Fischer, H. Freiesleben, D. Richter, S. Unholzer, Measurement and Analysis of Dose Rates and Gamma-Ray Fluxes in an ITER Shut-down Dose Rate Experiment, Fus. Eng. and Des. 63-64 (2002), 211-215

[8] H. Iida, D. Valenza, R. Plentada, R., Santoro, Radiation Shielding for ITER to allow for Hands-on Maintenance, J. Nucl. Sci. Techn,. Sup. 1 (March 2000),235-242.

[9] L. Petrizzi, P. Batistoni, U. Fischer, M. Loughlin, P. Pereslavtsev, R. Villari, Benchmarking of Monte Carlo Based Shutdown Dose Rate Calculations for Applications to JET, Radiation Protection Dosimetry 115(2005), 80-85.

[10] H. Tsige-Tamirat, On the use of CAD geometry for Monte Carlo particle transport, Proc. of the Monte Carlo 2000 conference, Lisbon 23-26 October 2000, p. 511.

[11] H. Tsige-Tamirat, U. Fischer, CAD Interface for Monte Carlo Particle Transport Codes. The Monte Carlo Method: Versatility Unbounded in a Dynamic Computing World ; Proc.of the Conf., Chattanooga, Tenn., April 17-21, 2005, La Grange Park, ANS, 2005

[12] H. Tsige-Tamirat, U. Fischer, P. Carman, M. Loughlin, Automatic generation of a JET 3D neutronics model from CAD geometry data for Monte Carlo calculations, Fus. Eng. Des. 75-79(2005), 891-895.

[13] H. Tsige-Tamirat, U. Fischer, A. Serikov, S. Stickel, Automatic Generation and Validation of an ITER Neutronics Model from CAD Data, 24th Symposium on Fusion Technology, Warsaw, Poland, September 11-15, 2006.

[14] S. P. Simakov, U. Fischer, U. von Möllendorff, I. Schmuck, A. Konobeev, P. Pereslavtsev, Advanced Monte Carlo Procedure for the D-Li Neutron Source Term Based on Evaluated Cross-Section Files, J. Nucl. Mat. 307-311 (2002), 1710-1714.

[15] A. Konobeev, Yu. Korovin, P. Pereslavtsev, U. Fischer, U. von Möllendorff, Development of Methods for Calculation of Deuteron-Lithium and Neutron-Lithium Cross Sections for Energies up to 50 MeV, Nucl. Sci. Eng. 139 (2001), 1.

[16] U. Fischer, S.P. Simakov, U. von Möllendorff, P. Pereslavtsev, P., Bem, P.P.H. Wilson, Validated Computational Tools and Data for IFMIF Neutronic Calculations, AccApp'03 ANS Embedded Topical Meeting, San Diego, CA, June 1-5, 2003.

[17] U. Fischer, M. Avrigeanu, P. Pereslavtsev, S.P. Simakov, I. Schmuck, Evaluation and Validation of D-Li Cross-Section Data for the IFMIF Neutron Source Term Simulation, 12th Int. Conf. on Fusion Reactor Materials (ICFRM-12), December 4 – 9, 2005, Santa Barbara, California, USA (to appear in J. Nucl. Materials).

[18] M. Hagiwara, M. Baba, N. Kawata, N. Hirabayashi, T. Itoga., Measurements of Neutron Emission Spectra and the 7Be Production in Li(d,n) and Be(d,n) for 25 and 40 MeV deuterons, JAERI Nuclear Data Symposium, December 2002.

[19] P.P.H. Wilson, Neutronics of the IFMIF neutron source: Development and Analysis. Report FZKA 6218, Karlsruhe 1999

[20] L. S. Waters (Ed), MCNPX User's Manual Version 2.1.5, available from http://mcnpx.lanl.gov.

[21] Y. Chen, Coupled Monte Carlo-Discrete Ordinates Computational Scheme for Three-Dimensional Shielding Calculations of Large and Complex Nuclear Facilities, Forschungszentrum Karlsruhe, FZKA 7075, April 2005

[22] Y. Chen, U. Fischer, Program system for three-dimensional coupled Monte Carlo-Deterministic shielding analysis with application to the accelerator-based IFMIF neutron source, Nuclear Instruments and Methods in Physics Research Section A, Volume 551, Issues 2-3 (2005), 387-395.

[23] DOORS 3.2, One-, Two- and Three-Dimensional Discrete Ordinates Neutron/Photon Transport Code System, RSICC Computer Code Collection, CCC-650, (1998).

[24] R. L. Perel, J. J. Wagschal, Y. Yevein, Monte Carlo Calculation of Point-Detector Sensitivities to Material Parameters, Nucl. Sci. Eng., 124 (1996),197-209.

[25] R. L. Perel, Monte Carlo Calculation of Sensitivities to Secondaries' Angular Distributions, Nucl. Sci. Eng. 143 (2003), 121-131.

[26] D. Leichtle, U. Fischer, I. Kodeli, R. L. Perel et al, Sensitivity and Uncertainty Analyses of the Tritium Production in the HCPB Breeder Blanket Mock-up Experiment, 24th Symposium on Fusion Technology, Warsaw, Poland, September 11-15, 2006.

[27] U. Fischer, P. Batistoni, E. Cheng, R. A. Forrest, T. Nishitani, Nuclear Data for Fusion Energy Technologies: Requests, Status and Development Needs, Int. Conf, on Nuclear Data for Science and Technology, Santa Fe, NM, September 26-October 1, 2004, AIP Conference Proceedings Vol. 769, Melville New York 2005, pp 1478-1485.

[28] D. Muir, S. Ganesan, FENDL – a reference nuclear data library for fusion applications, Int. Conf. Nucl. Dat. Sci. Techn., Jülich, May 13-17, 1991.

[29] D. L. Aldama, A.Trkov, FENDL-2.1: Update of an evaluated nuclear data library for fusion applications, Report INDC(NDS)-467, December 2004.

[30] A.J. Koning, R. Jacqmin, R. Forrest, O. Bersillon, P. Rullhusen, A. Nouri and M. Kellett, Status of the JEFF nuclear data library, AIP Conf Proc. 769 (2005) 177, Int. Conf. on Nuclear Data for Science and Technology, Santa Fe, 26. Sept. - 1. Oct. 2004, USA, and NEA Data Bank, JEFF Report 19 (to be published)

[31] P. Pereslavtsev, U. Fischer, Evaluation of n + W cross section data up to 150 MeV neutron energy, AIP Conf Proc. 769 (2005) 177, Int. Conf. on Nuclear Data for Science and Technology, Santa Fe, 26. Sept. - 1. Oct. 2004, USA.

[32] P. Pereslavtsev, U. Fischer, Neutron Cross Section Data Evaluation for Ta-181 up to 150 MeV, Nuclear Inst. and Methods in Physics Research, B, Vol 248/2 , 225-241

[33] P. Batistoni et al., Neutronics experiment on a HCPB breeder blanket mock-up, 24th Symposium on Fusion Technology, Warsaw, Poland, September 11-15, 2006.