

MONTE CARLO APPLICATIONS IN FUSION TECHNOLOGY

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ABSTRACT

An overview is presented of Monte Carlo applications in fusion technology conducted at Forschungszentrum Karlsruhe in the framework of the European Fusion Technology Programme. The focus is on applications which require dedicated methods and computational tools to satisfy specific needs in the design and analysis of fusion devices such as ITER, the International Thermonuclear Experimental Reactor, JET, the Joint European Torus, and IFMIF, the International Fusion Material Irradiation Facility.

For applications to ITER and JET, a programme system has been developed which is suitable for the calculation of Monte Carlo based shut-down dose rate distributions in full 3D geometry. The system makes use of the MCNP code for the transport calculations (neutron and decay photons) and the FISPACT inventory code for the activation calculations with a coupling scheme for the automated routing of decay photon source distributions and neutron flux spectra. For neutronics analyses of the IFMIF D-Li neutron source, a dedicated Monte Carlo based computational approach has been developed. The resulting McDeLicious code is an enhancement to MCNP with the new ability to sample in the transport calculation the generation of d-Li source neutrons on the basis of tabulated $d + 6,7\text{Li}$ cross-section data. For IFMIF shielding calculations, a mapping approach was developed to couple full 3D McDeLicious calculations with the discrete ordinates procedure for deep penetration calculations using the TORT code. For the efficient use of CAD geometry data in Monte Carlo calculations, suitable conversion algorithms have been developed and implemented into an interface programme with a graphical user interface.

In the paper the methods and tools mentioned are reviewed, results of validation analyses are presented and various applications examples are given.

Key Words: fusion technology, Monte Carlo, neutronics, IFMIF, ITER, JET, CAD interface

1. INTRODUCTION

The Monte Carlo method has proven to be the most suitable 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 modeled 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 is thus only affected 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, the International Thermonuclear Experimental Reactor, JET, the Joint European Torus, and IFMIF, the International Fusion Material Irradiation Facility, dedicated methods and computational tools are required though.

This paper presents an overview of Monte Carlo applications in fusion technology conducted at Forschungszentrum Karlsruhe in the framework of the European Fusion Technology Programme. The focus is on the application and validation of computational methods and tools developed recently for nuclear analyses of ITER and other like fusion devices

as well as the IFMIF neutron source facility. These include a Monte Carlo based computational scheme for the calculation of three-dimensional shut-down dose rate distributions, algorithms and interfaces for the conversion of CAD models into semi-algebraic geometry representation for use in Monte Carlo transport calculations as well as computational techniques and data for IFMIF neutronics and activation calculations. In the following these methods and tools are described, results of validation analyses are presented and various applications examples are given.

2. MONTE CARLO BASED 3D SHUT-DOWN DOSE RATE CALCULATIONS

During D-T operation of experimental fusion devices such as JET, the Joint European Torus, and, in the near future ITER, machine components are activated by neutron radiation generated in the plasma chamber. For safe operation and maintenance of the 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, has been recently developed for shutdown dose rate analyses of ITER [1]. The R2S system is based on the use of the MCNP [2] transport and the FISPACT [3] inventory code linked through a suitable coupling scheme for the automated routing of decay gamma source and neutron flux spectrum distributions. The decay gamma source distribution is sampled in a source routine linked to MCNP according to the distribution provided by preceding MCNP neutron transport and FISPACT inventory calculations.

The shutdown dose rate is due to decay photons emitted by radioactive nuclides generated during irradiation. The calculation of shutdown dose rate distributions, therefore, requires first a neutron transport calculation for the spatial distribution of the neutron flux spectra, second a nuclide inventory calculation for the decay gamma source distribution, and third a decay gamma transport calculations for the dose rate distribution. The transport calculations, both for neutrons and decay gammas, are performed with the Monte Carlo code MCNP to enable a proper geometrical representation. This is essential when dealing with complex systems like the JET and ITER devices. The activation calculation is performed with the FISPACT inventory code using FENDL-2.0/A activation cross-section data [4].

The Monte Carlo code MCNP and the activation inventory code FISPACT thus form the core of the computational scheme. They are linked through interfaces for the automated routing of the neutron flux spectra from MCNP to FISPACT (interface MCFISP) and the decay gamma source distribution from FISPACT to MCNP (interface FISPMC) as shown in Fig. 1.

Neutron flux spectra are calculated by MCNP for all non-void cells that may be activated and result in decay gamma sources after irradiation. The spectra are stored in the standard way in the MCNP tally file MCTAL. The interface MCFISP passes the neutron flux spectrum of a specified geometry cell to FISPACT for the material activation calculation. Thus FISPACT provides the activation inventory and the decay gamma source (intensity and spectrum) for the material and geometry cell under consideration. This procedure is automatically repeated for all geometry cells specified in the MCFISP input file.

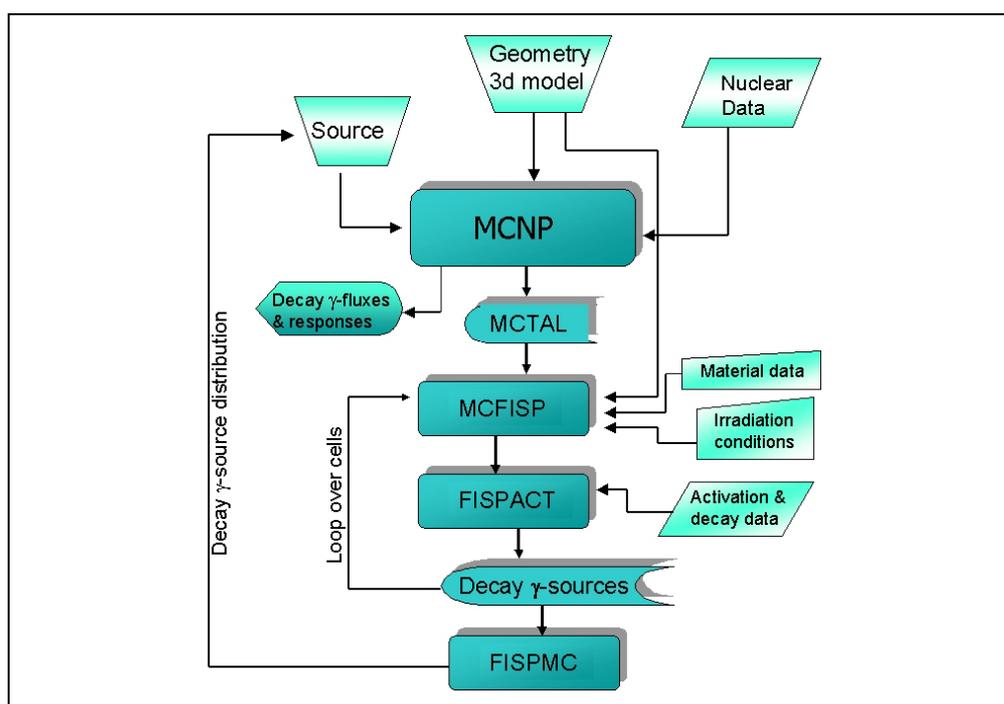


Fig. 1: Flow scheme for MCNP based 3D shut-down dose rate calculations

The interface FISPMC reads the decay gamma source files generated for all of the specified geometry cells and creates one single decay gamma source file that contains the decay gamma source intensities, the spectra, the cell volumes and the cell identifications for all considered geometry cells and cooling times. The decay gamma transport calculation is again performed with MCNP using the geometry model of the neutron transport calculation. The decay gamma source distribution is sampled in a FORTRAN subroutine linked to MCNP by making use of the decay gamma source file provided by the FISPMC interface. Only one single source sampling volume need to be specified by defining its boundaries in the MCNP input deck. The source sampling volume may cover the complete MCNP geometry model or only parts there-of which are relevant to the shutdown dose rate to be calculated. For generating a decay gamma, a point is sampled uniformly in space within the specified source sampling volume. Next the decay source intensity of the associated cell is assigned to the source gamma as its weight. In this way, the sampled source distribution represents the decay gamma source distribution as provided by the preceding MCNP/FISPACT calculations. This approach enables the programme system to handle any complex geometry model as it would not be possible in the conventional way by specifying source sampling volumes on MCNP's source definition card.

The ITER shutdown dose rate experiment conducted at the ENEA Frascati Neutron generator FNG [5] was used as a first experimental benchmark to validate the programme system. 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 [6]. Calculated and measured shutdown dose rates are compared in Fig. 2. An overall satisfactory agreement was obtained over the considered range of decay times. The programme system was thus considered as qualified for application to ITER and JET.

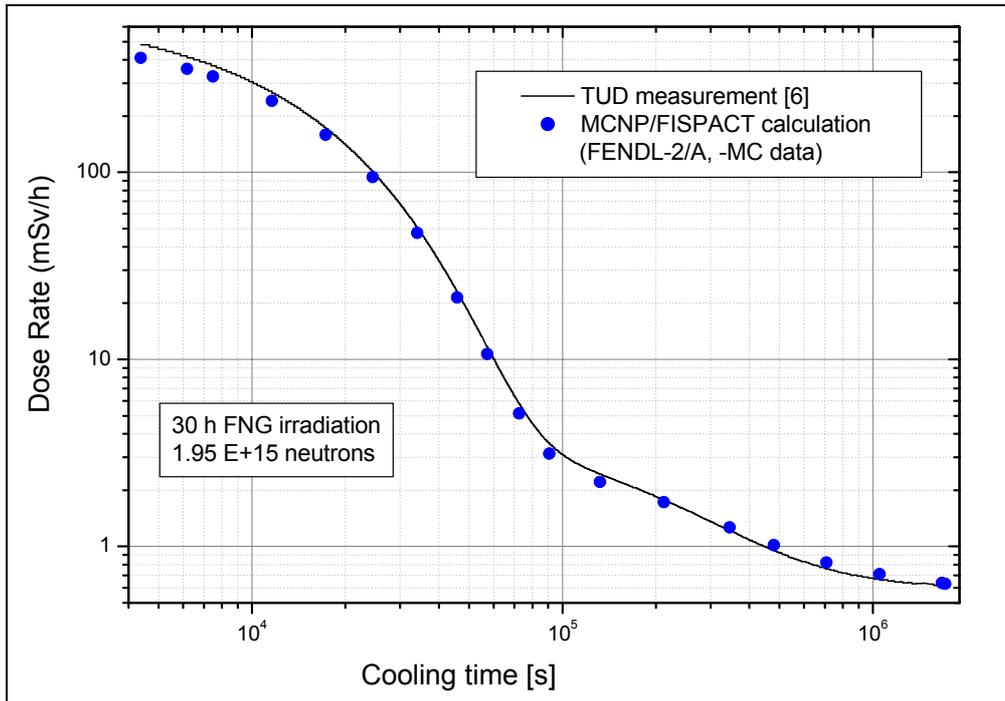


Fig. 2: ITER shutdown dose rate experiment at the Frascati Neutron Generator (FNG): Measured and calculated shutdown dose rates

As a first proof-of-principle, the R2S programme system was applied to the shut-down dose rate analysis of the ITER mid-plane maintenance port section making use of 20° torus sector MCNP model [1]. This application demonstrated that full 3D shutdown dose rate calculations can be performed for large and complex reactor systems. Care must be taken, however, to ensure the spatial resolution is fine enough for a good representation of the neutron flux gradient in regions where important decay gamma sources are located. It is therefore required to refine the spatial segmentation of MCNP models developed for neutron transport calculations.

As an important next step, the R2S programme system was applied to JET to test its capability for shutdown dose rate calculations of an existing complex experimental fusion device. Benchmark tests on JET were first made by means of comparison calculations with the so-called direct one step (D1S) approach [7] employing a 3D torus sector model of the device. 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 $\pm 25\%$ [8]. 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, see Fig. 3 a) 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. 3 b - d. The comparison between calculational and experimental JET dose rate results was constrained, however, by the rather high uncertainties associated with the available measurements. These data were recorded by the JET Health Physics team as part of the regular monitoring programme under not well defined conditions and were judged to be not accurate enough to validate computational results. It was therefore concluded that a dedicated experiment on JET need to be conducted for the benchmarking of R2S and D1S shut-down dose rate calculations. This is scheduled for the JET 2005 experimental programme during D-D operation.

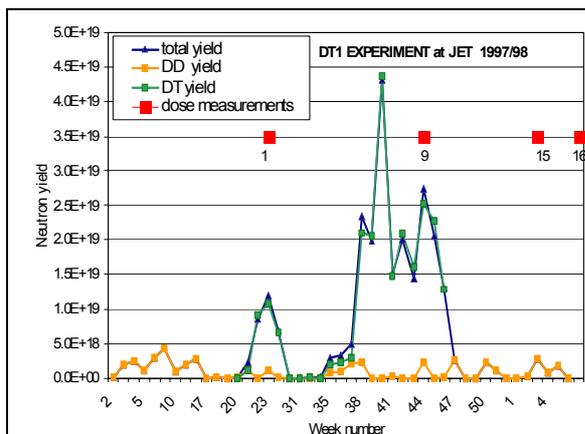


Fig. 3a: JET DTE 1 neutron yields and dose measurements time points (red squares).

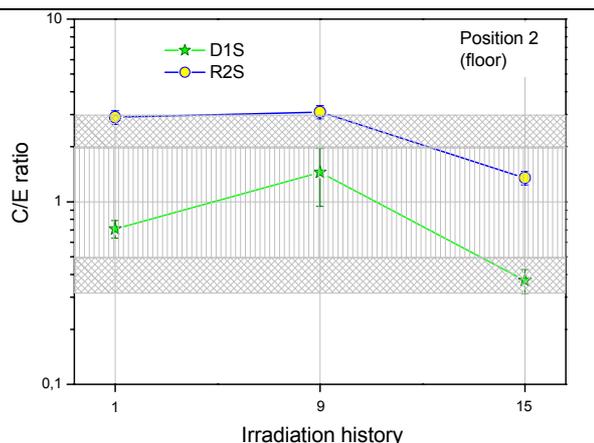


Fig. 3b: C/E ratios for JET position 2 (on the torus hall floor).

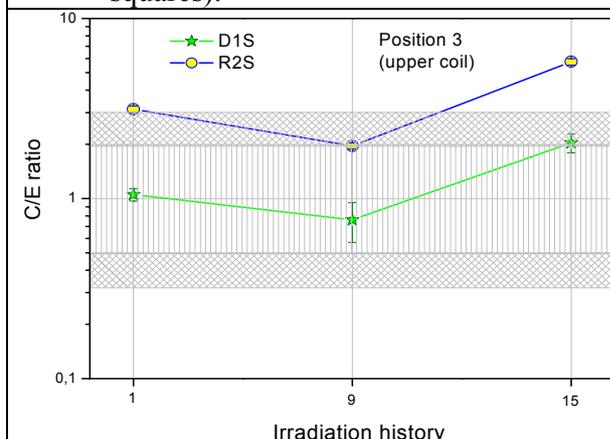


Fig.3c: C/E ratio of dose rates for JET position 3 (in contact with the upper coil).

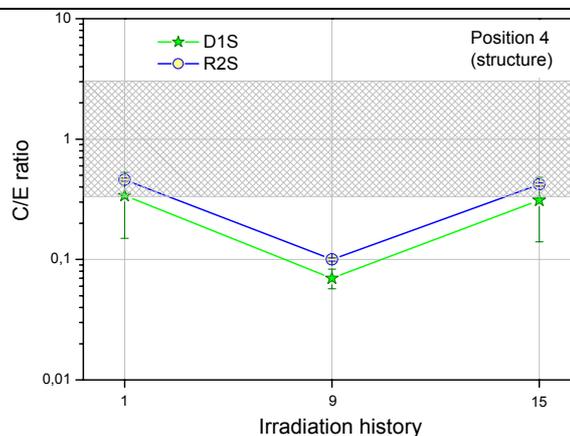


Fig.3d: C/E ratio dose rates for JET position 4 (in contact with the machine structure).

Fig. 3: Neutron yields and C/E ratios of calculated (C) and experimental (E) dose rates for different irradiation histories and positions of JET following the DTE1 experimental campaign.

3. D-LI NEUTRON SOURCE GENERATION AND TRANSPORT SIMULATION

The IFMIF neutron source uses the d-Li stripping reaction to produce neutrons for high fluence irradiations of fusion power reactor candidate materials. A flowing liquid lithium target is bombarded by high current deuteron beams accelerated up to 40 MeV energy. The resultant neutron spectrum is fusion-relevant but includes a high-energy tail that extends up to 55 MeV neutron energy. Dedicated computational tools and nuclear data have been developed over the past years for IFMIF neutronics and activation calculations [9,10].

The McDeLi code [11] has been previously developed as an extension to MCNP with the capability of representing the D-Li neutron source term on the basis of a built-in semi-empirical d-Li reaction model. McDeLi can handle two beams impinging onto the lithium target taking into account different beam directions and a spatially varying intensity distribution. Deuteron slowing down in the lithium is described according to the well established empirical model of J. Ziegler et al. [12]. The Li(d,xn) reaction model considers as neutron producing reactions deuteron stripping and deuteron absorption followed by the formation of a compound nucleus with subsequent neutron emission. Adjustable parameters of the Li(d,xn) reaction model were obtained through numerical fits to experimental angle-energy distributions of neutron yields from thick lithium targets bombarded by 32 and 40 MeV deuterons. Extensive testing of the McDeLi code against available experimental data over the full deuteron energy range from 5 to 50 MeV has shown that the experimental data below 30 MeV incident deuteron energy cannot be reproduced. The high energy tail above 40 MeV cannot be represented either since the semi-empirical reaction model does not take into account exothermic reactions.

The McDeLicious Monte Carlo code [13] was developed more recently with the objective to simulate in the transport calculation the neutron generation on the basis of evaluated d + $^{6,7}\text{Li}$ cross-section data. A complete set of evaluated cross-section data was prepared to this end for the reaction system d + $^{6,7}\text{Li}$ [14]. The evaluated data include cross-sections for all reaction channels up to 50 MeV deuteron energy as well as energy-angle distributions for the neutrons emitted through the various $^{6,7}\text{Li}(d,xn)$ -reactions. The cross-section data set was prepared in standard ENDF-6 format and processed with the ACER module of the NJOY99 code [15]. With these data available, the generation of D-Li source neutrons can be sampled in the Monte Carlo calculation. First the deuteron track length is sampled taking into account the total d-Li interaction cross sections. Next the interaction probability with either of the ^6Li or ^7Li nuclides is calculated according to the associated macroscopic cross sections for the given deuteron energy. Eventually the energy and angle of the generated neutron and photon are sampled. This information together with the coordinates of the d-Li interaction in the lithium target cell is further used for transporting the generated neutrons and photons in the Monte Carlo calculation. This procedure has been integrated to the McDeLi code by replacing the routines describing its semi-empirical Li(d,xn) reaction model. The resulting McDeLicious Monte Carlo code is thus a further enhancement to McDeLi with the new ability to sample the generation of d-Li source neutrons and photons from tabulated d + $^{6,7}\text{Li}$ cross-section data as provided on the ACE formatted data file.

The McDeLicious approach was extensively tested against available experimental thick lithium target neutron yield data. Figs. 4 a and b show comparisons of measured and calculated neutron yields as function of the deuteron incidence energy. There are included calculation results obtained with the semi-empirical d-Li reaction model of McDeLi and the ISABEL intra-

nuclear cascade model of the high energy particle Monte Carlo code MCNPX 2.1.5 [16]. It is revealed that McDeLicious can predict the D-Li neutron generation with the best achievable accuracy over the whole deuteron energy range from threshold up to 40 MeV. MCNPX, on the other hand, shows a clear trend for underestimating the neutron yields.

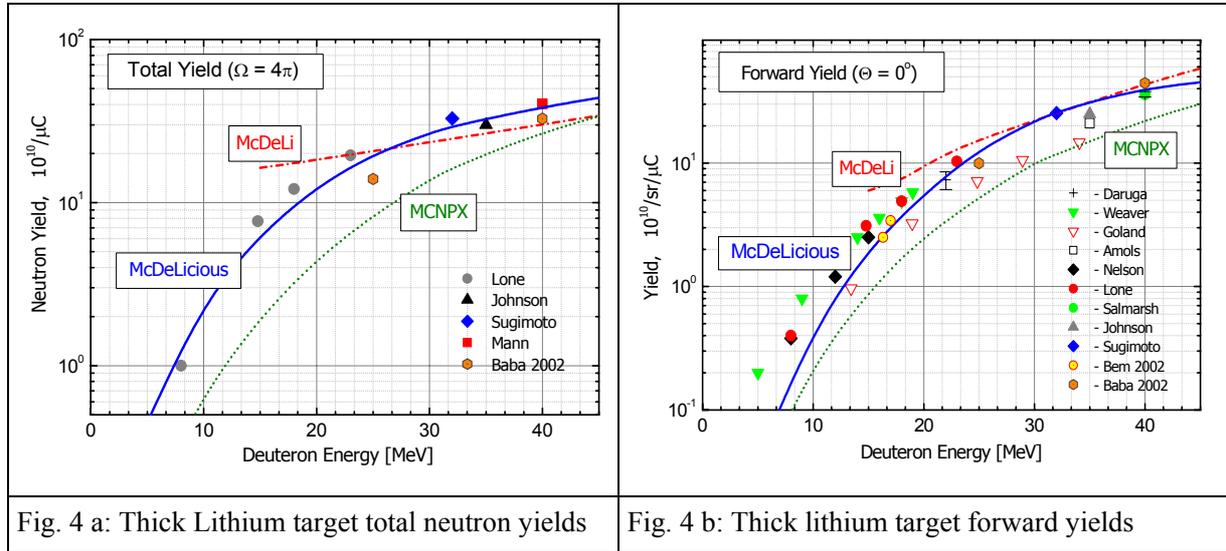


Fig. 4: Comparison of calculated and measured thick lithium target neutron yields as function of the deuteron incidence energy.

Testing of the McDeLicious approach against double differential thick target neutron yields showed an overall good agreement while MCNPX calculations tend to underestimate the neutron spectra at forward angles, see Figs. 5 a-d. The recent experimental data of Baba et al. [17] at 25 and 40 MeV deuteron energies indicate, however, that McDeLicious overestimates the production of neutrons with energies less than 2 MeV. This is due to the fact that the experimental Lone et al. data [18] have been used in evaluating the $d + {}^{6,7}\text{Li}$ cross-section data. These data show a rather large neutron production in the low energy range below 2 MeV which is in contradiction to other measurements including the recent Baba et al. time-of-flight data. Both the recent Baba et al. and the Bem et al. [19] experiments for thin lithium target yield spectra indicate also the need to re-evaluate the data for the $d + {}^7\text{Li}$ reaction channels populating the excitation states of the residual ${}^8\text{Be}$ nuclei. This will affect the high energy tail of the neutron yield spectra.

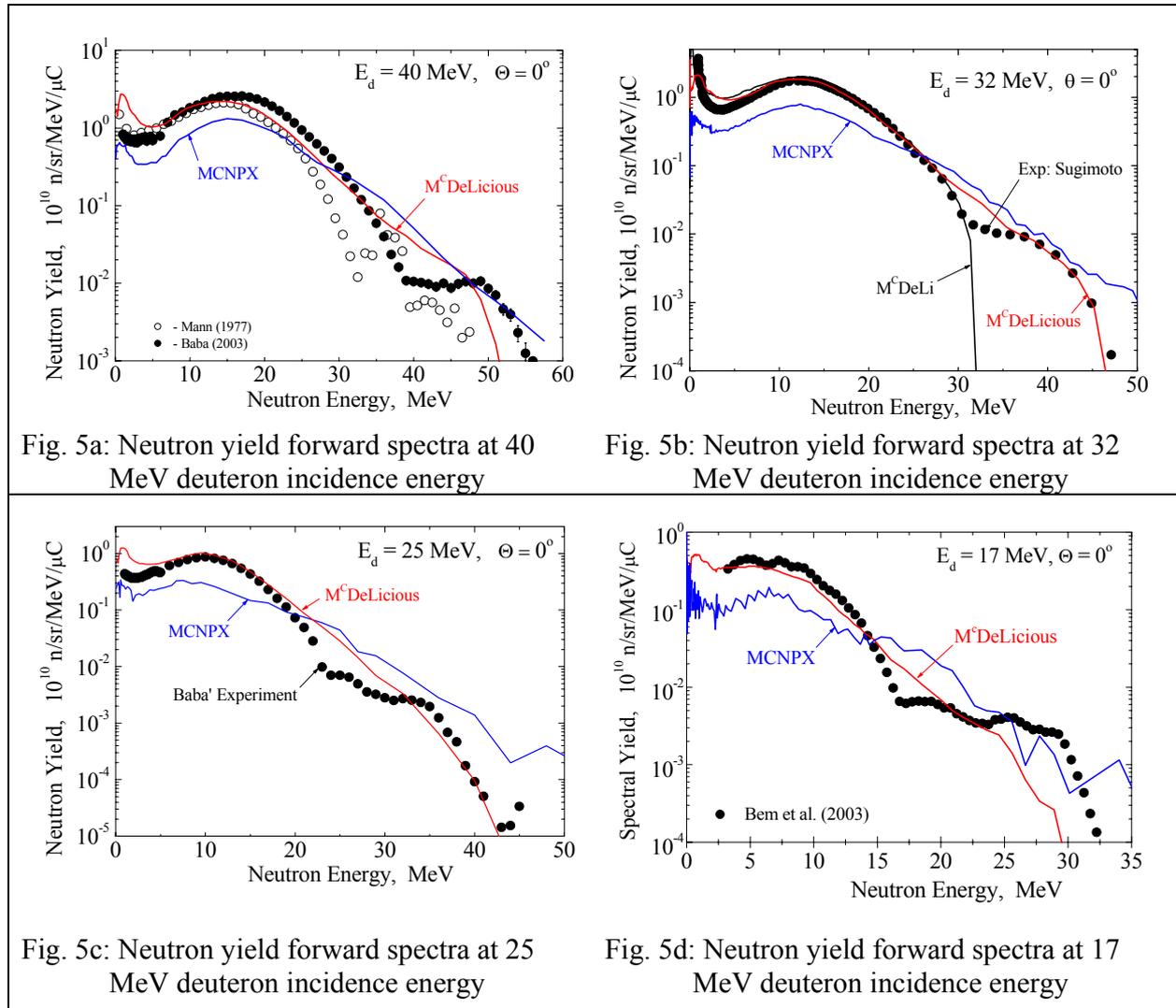


Fig. 5: Comparison of calculated and measured thick lithium target neutron yield forward energy spectra.

The primary mission of the IFMIF neutron source facility is to generate a materials irradiation database for the design, construction, licensing and operation of a fusion power demonstration reactor (DEMO). The major neutronics tasks in this context are (i) to prove the suitability of IFMIF as a neutron source to properly simulate fusion-like irradiations of reactor materials and (ii) to provide the data required for the design and optimization of the irradiation test modules and the lay-out of the test cell. These include neutron/photon transport calculations to obtain the flux distributions and nuclear responses such as the nuclear heating, the radiation damage accumulation and the gas production. Another important neutronics issue is the activation of the test cell components and the elemental transmutations of the materials irradiated in the test modules.

Using the McDeLicious code with high energy cross-section data from various sources, extensive neutronics analyses have been previously performed for the characterization of the High Flux Module (HFTM) of IFMIF using simplified geometry models, see e. g. Ref. [20]. A detailed and comprehensive 3D model of the IFMIF test cell has been recently developed on the basis of the latest CAD design [21] including the deuteron beam pipes, the lithium target with the back plate (BP), the lithium loop components inside the test cell, the vertical test assemblies accommodating the high and low flux test modules (HFTM, LFTM), the universal testing machine (UTM), a tungsten spectral shifter plate (W), the tritium release module (TRM), a carbon reflector (C), and, finally the cover, walls and floor of the test cell, see Fig. 6 for a 2D view (vertical cut) of the Monte Carlo calculation model.

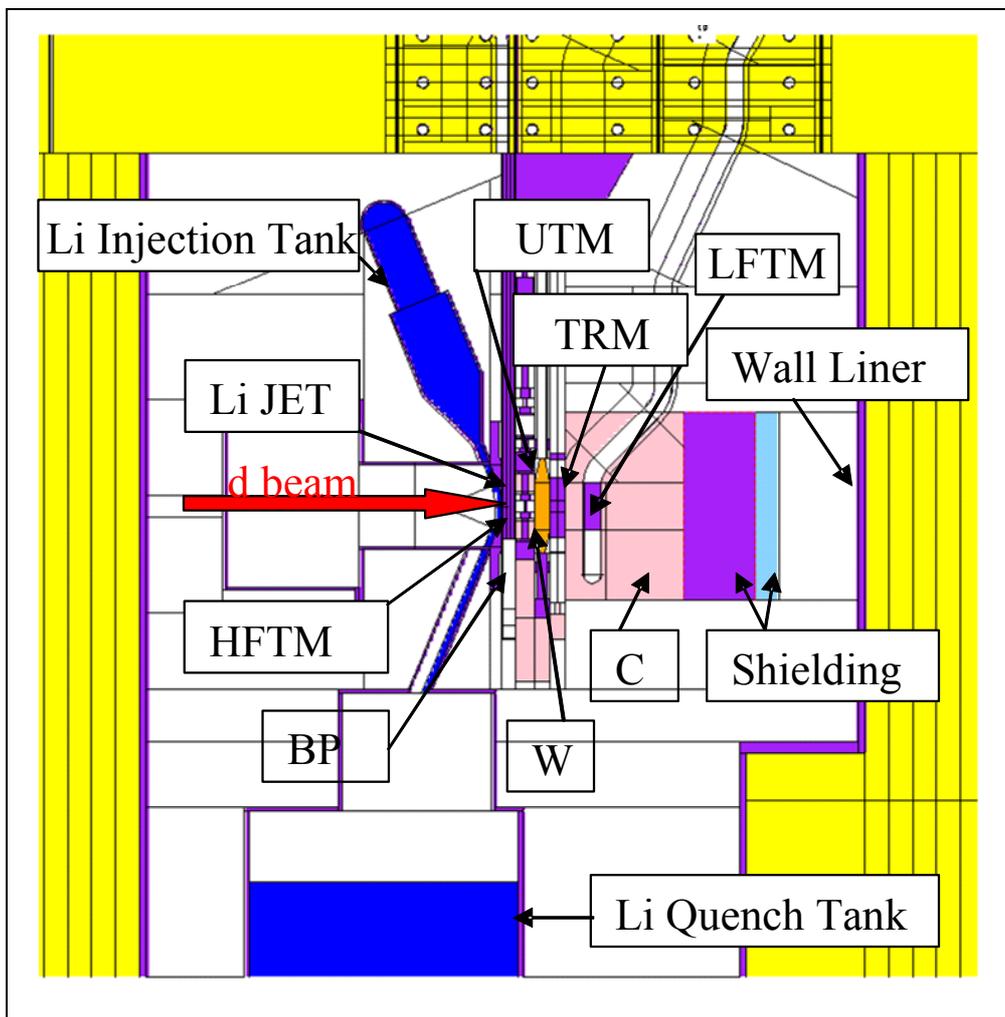


Fig. 6: MCMP model of IFMIF test cell with major components indicated

Comprehensive neutronics and activation calculations have been performed for the entire IFMIF test cell applying the McDeLicious Monte Carlo code for the transport calculations and the ALARA inventory code [22] with IAEA-2001 cross-section data [23] for the activation

calculations. Neutron flux distributions, displacement damage, gas production and heating rates, the induced radioactivity inventories as well as the resulting contact γ -dose rates have been assessed for the Li target back plate, the Test Modules, the Universal Testing Machine, the Tritium Release and the Low Flux Test Modules [24]. As an example, Fig. 7 shows contour plots of the displacement damage and gas production rate distributions calculated for the high flux test modules with irradiation specimens made of the reduced activation ferritic-martensitic steel Eurofer.

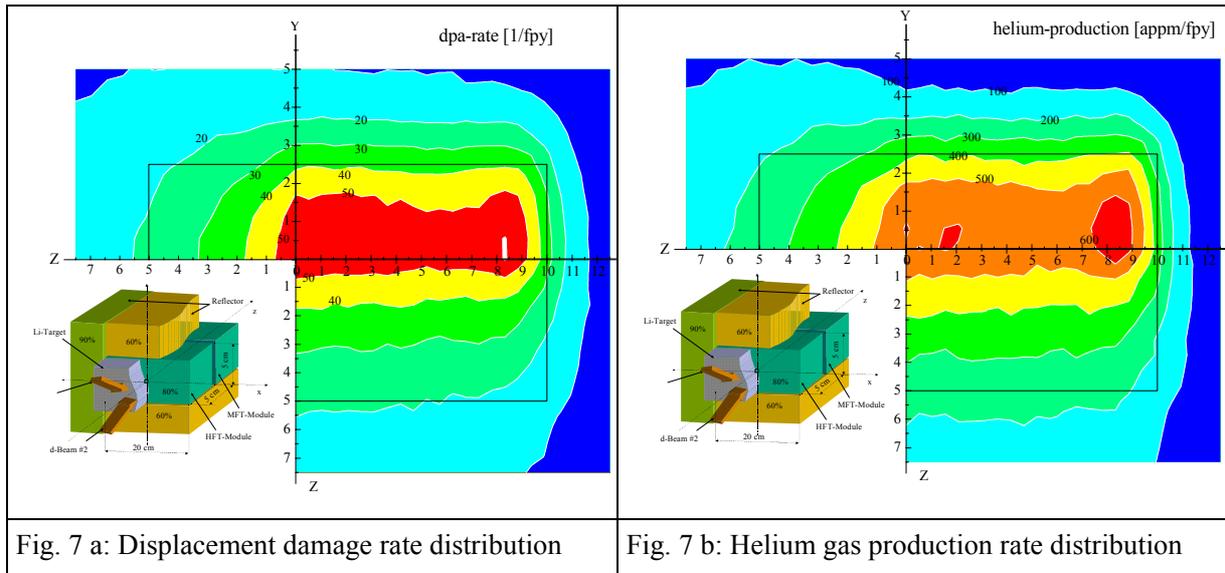


Fig. 7: Contour plots of material relevant nuclear responses in the IFMIF high flux test module with Eurofer specimens.

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. The deep penetration of radiation through bulk shields is a challenge for the Monte Carlo particle transport simulation while approximations are required to model complex geometries by the discrete ordinates method. To better handle such kinds of shielding problems, a dedicated computational approach for coupled Monte Carlo – deterministic transport calculations has been developed [25]. The Monte Carlo technique is used to simulate the particle generation and transport in and around the neutron source region (involving complex geometries) and 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 [26], see the flow chart (Fig. 8 a). Use is made of MCNP's surface source write (SSW) feature providing a complete set of data for the individual particle tracks including the exact position and

direction vectors, their energies and weights. The whole set of data recorded on the SSW binary file is processed by the interface programme to generate a boundary source file for discrete ordinates calculations with the TORT code by making use of the mapping technique.

For IFMIF shielding calculations, this approach enables 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 $d + \text{Li}$ cross-section data [26]. The neutron transport through the thick concrete walls surrounding the Test Cell is described by means of 3D S_N calculations with the TORT code using the boundary source distribution calculated by McDeLicious at the inner surface of the Test Cell wall. 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. 8b). 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 reference shielding material of the test cell walls was assumed to be heavy concrete.

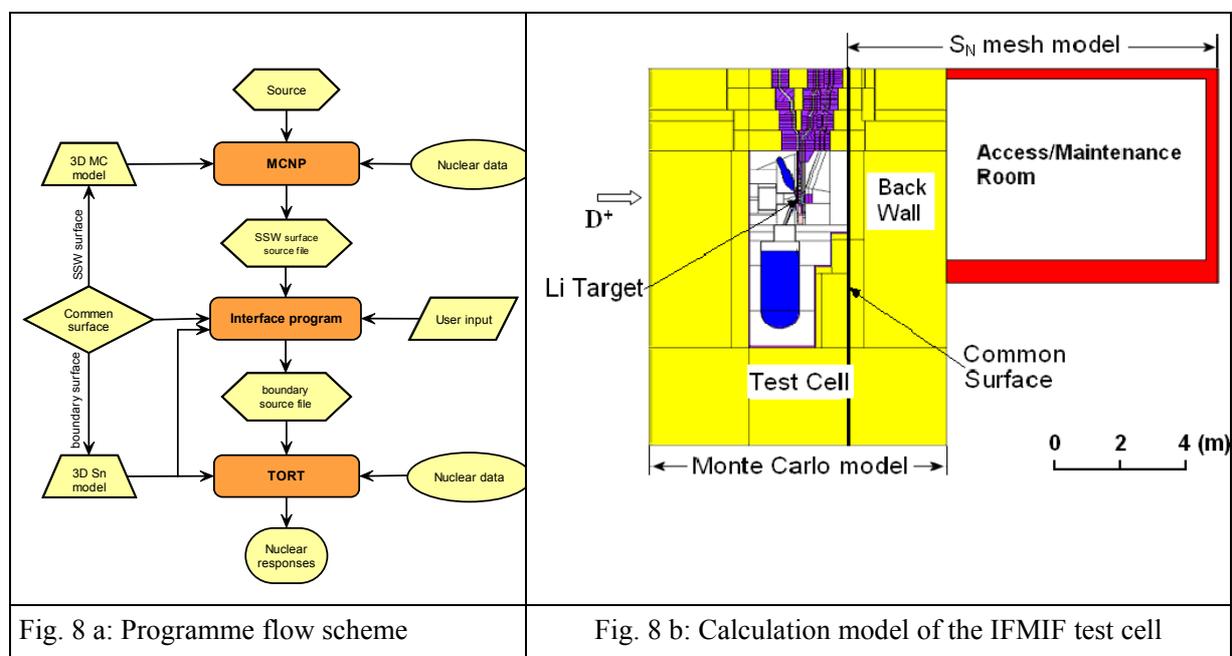


Fig. 8 : Computational scheme and geometry model for coupled Monte Carlo/ S_N transport calculations of the IFMIF test cell.

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 and multiplying them by the flux-to-dose rate conversion factors according to the ICRP specifications. Fig. 9 a shows the dose rates as a function of the distance from the inner surface of the back wall. It is seen that heavy concrete provides a significant better shielding performance than ordinary concrete. The dose rate attenuates across the back wall by about 9 orders of magnitude in the case of heavy concrete and about 8 orders of magnitude in the case of ordinary concrete. In the access/maintenance room the dose rate decreases by a factor of

about 25 along the beam direction in both cases. It is also found that the dose rate is dominated by neutrons while photons contribute less than 15%.

The peak dose rate in the access/maintenance room exceeds the design limit of $10\mu\text{Sv/h}$ by about 3 orders of magnitude for ordinary concrete and by about 2 orders of magnitude for heavy concrete. To reduce the dose rate level in the room to the design limit, the thickness of the back wall would have to be increased to about 400 cm in the case of heavy concrete and to about 450 cm in the case of ordinary concrete. Fig. 9 b shows the 3-D dose rate distribution calculated for the access/maintenance room with heavy concrete as shielding material. It is noted that the profiles of the dose rates are still very similar to the incident beam profile.

Significant discrepancies were found when comparing the results of the coupled MC/S_N calculation to those of previous IFMIF shielding calculations using an approximate neutron source representation and a simplified one-dimensional geometrical model. This indicates that a suitable and qualified computational tool such as the coupled MC/S_N scheme is required for reliable shielding analyses of IFMIF.

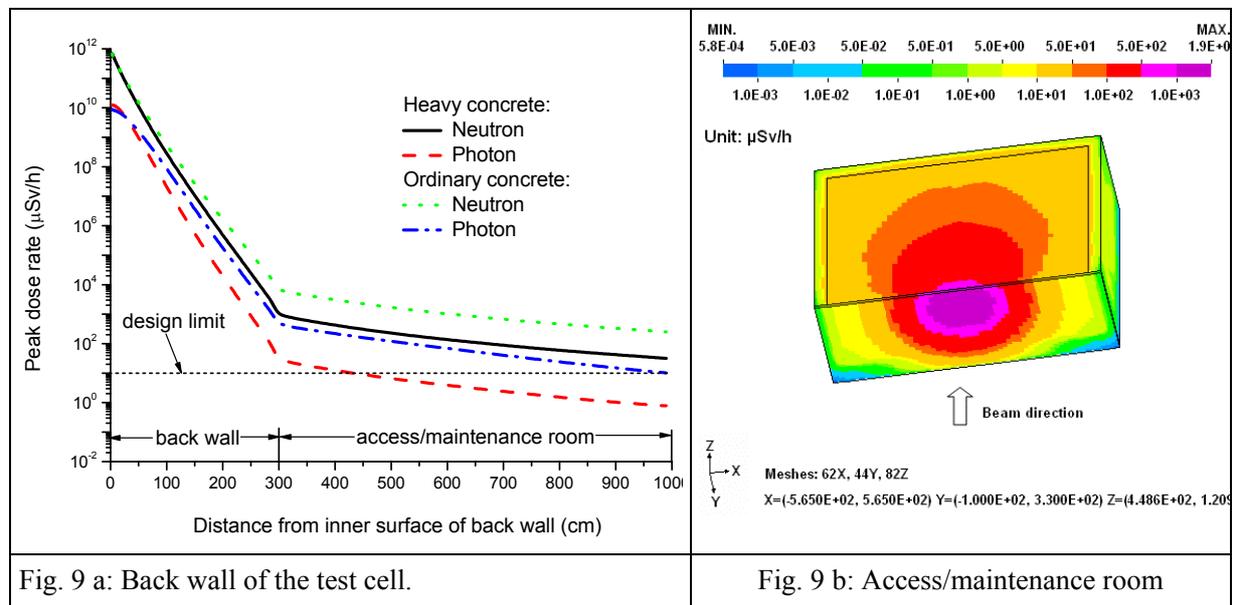


Fig. 9: Dose rate distributions calculated with the coupled MC/S_N coupling scheme for the IFMIF test cell at full power operation.

5. 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 modeling 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 [27] and have been implemented into a first version of an interface programme with a graphical user interface based on a CAD kernel and graphics software. The development of a user friendly interface programme with advanced features is currently underway [28].

The geometry needed for the MC particle transport simulation is a decomposition of the problem space into a finite collection of disjoint regions (cells) whose union is the problem space. It is usually represented as a Boolean form of primitive solids or algebraic half-spaces. Most commercial CAD systems use the boundary representation (B-rep) method to store geometric models of solids. A solid is assumed to be a compact and regular point set, whose boundary is composed of closed oriented manifolds. The difference in the representation schemes makes a conversion necessary. The geometric and topological data of a B-rep solid from a CAD system is used to construct its semi-algebraic representation as employed in MC codes. The two main steps involved in this approach are the access to the data of the CAD system and the conversion process itself. Access to the data of a CAD system can be provided through export of the data in neutral format. Given a solid in boundary representation, it can be shown that its semi-algebraic representation is computable. The conversion proceeds in two steps. After the traversal of the B-rep data structure of a solid and extraction of the boundary supports, the first step is to determine the definability of the solid by the available boundary support set. If this is not the case, the set is enlarged until definability is achieved. In the second step, a cell construction is performed by a sign constant decomposition of the solid by the boundary supports.

The interface programme developed for the conversion of the CAD data into the semi-algebraic surface representation is based on a software design integrating a CAD kernel, which is a C++ class library, a graphical user interface (GUI) and the conversion algorithm. The C++ based GUI is used for the 3D visualization with the capability of manipulating the geometry model. Standard CAD interface files (IGES and STEP format) can be imported and converted to the MCNP geometry representation. The conversion algorithm relies on the CAD kernel for its geometric and related computations. The implementation of the interface program is realized in a framework like library. The automatic generation of the MCNP geometry representation is treated as data exchange operation.

A first successful test application has been recently performed for a full octant model of the JET tokamak [29]. Starting from available design models, a suitable CAD model of JET octant 3 was generated at the JET drawing office using CATIA V5. Several iteration steps were required to optimise the CAD model for the conversion and the use with MCNP. Free form surfaces (B-Splines) e. g. can neither be treated by the interface nor used in MCNP calculations and had to be replaced by algebraic surfaces. The original model also included lots of details which are unsuitable for neutronic calculations and had to be discarded. Data files in STEP API-214 format were used to exchange the geometry data between the CATIA system and the interface programme.

The final CAD model as used for the conversion is shown in Fig. 10 a. It consists of 83 solids with 696 surfaces (517 planes, 140 cylinders, 31 cones and 7 tori). The converted MCNP model of the JET torus sector is shown in Fig. 10 b. It is composed of 210 geometry cells with 1386 surfaces (1116 planes, 214 cylinders, 36 cones and 20 tori) generated automatically by the interface programme.

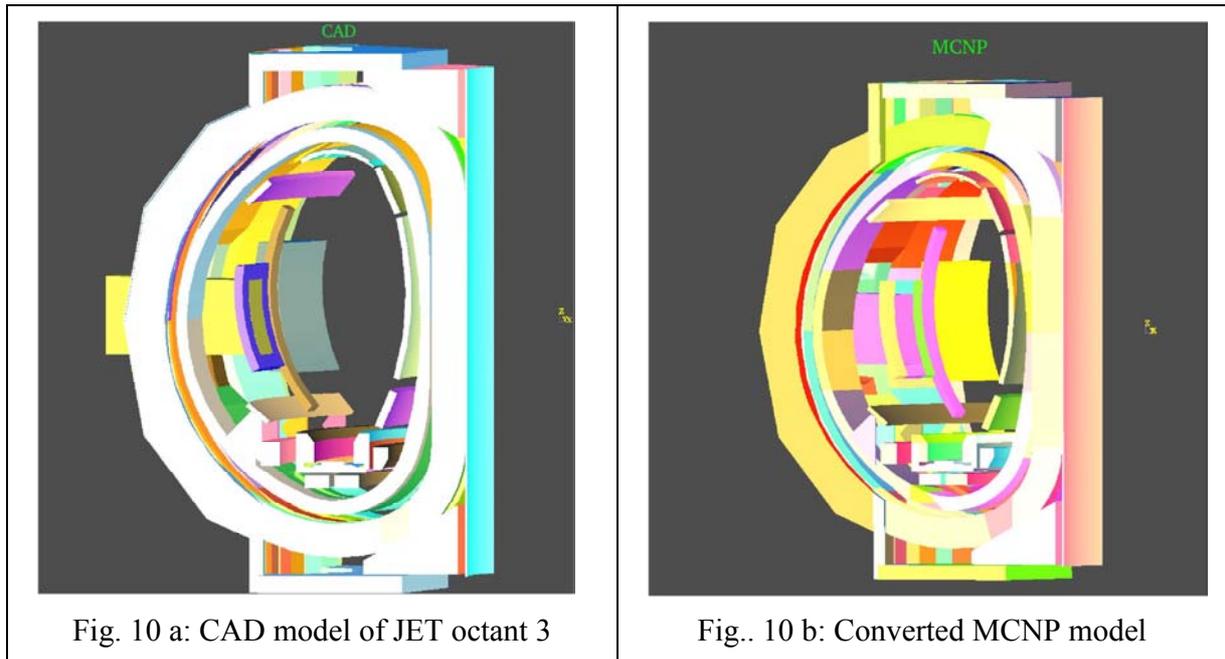


Fig. 10: Comparison of CAD and MCNP geometry models converted by the CAD-MCNP interface programme.

For validating the converted geometry model, stochastic volume calculations of parts were performed with MCNP showing very satisfactory agreement with the volumes provided by CATIA. It was thus concluded that the automatic conversion process reproduces the CAD geometry properly.

6. CONCLUSIONS

An overview has been presented of Monte Carlo applications in fusion technology conducted at Forschungszentrum Karlsruhe with the focus on applications requiring dedicated methods and computational tools for the design and analysis of fusion related devices. These include a Monte Carlo based computational scheme for the calculation of three-dimensional shut-down dose rate distributions, algorithms and interfaces for the conversion of CAD models into semi-algebraic geometry representation for use in Monte Carlo transport calculations as well as computational techniques and data for IFMIF neutronics and activation calculations. The methods and tools were described, results of validation analyses were presented and various applications examples given.

7. ACKNOWLEDGMENTS

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