EUROFUSION WPJET3-PR(16) 15870

A Cufar et al.

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Preprint of Paper to be submitted for publication in
Nuclear Instruments and Methods in Physics Research
Section A

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

Calculations to support JET neutron yield calibration: Modelling of neutron emission from a compact DT neutron generator

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Abstract

At the Joint European Torus (JET) the ex-vessel fission chambers and in-vessel activation detectors are used as the neutron production rate and neutron yield monitors respectively. In order to ensure that these detectors produce measurements that are accurate in absolute terms they need to be experimentally calibrated. A new calibration of neutron detectors to 14 MeV neutrons, resulting from deuterium-tritium (DT) plasmas, is planned at JET using a compact accelerator based neutron generator (NG) in which a D/T beam impinges on a solid target containing T/D, producing neutrons by DT fusion reactions. The present paper describes the analysis that was performed to accurately model the neutron source characteristics in terms of energy spectrum, angle-energy distribution and the effect of the neutron generator geometry. Different codes capable of simulating the accelerator based DT neutron sources are compared and sensitivities to uncertainties in the generator's internal structure analysed.

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

Preprint submitted to Nuclear Instruments and Methods in Physics Research Section A April 21, 2016

The analysis was performed to support preparation to the experimental measurements performed to characterise the NG as a calibration source. Further extensive neutronics analyses, performed with this model of the NG, will be needed to support the neutron calibration experiments and take into account various differences between the calibration experiment and experiments using the plasma as a source of neutrons.

Keywords: compact DT neutron generator, MCNP, modelling

1. Introduction

1.1. Calibration of neutron detectors

At the Joint European Torus (JET), neutron detectors, namely fission chambers located near the transformer limbs on the outside of the vacuum vessel and activation system located in the vacuum vessel near the plasma are used as main neutron emission rate and neutron yield monitors respectively [1]. As neutron emission is linearly proportional to the produced fusion power, these detectors are also used to determine the absolute fusion power during operation and energy released in a discharge. In order for these detectors to measure the neutron emission in absolute terms they need to be calibrated to a source with known characteristics – a calibration source. This source must be well characterised in order to minimise the experimental uncertainty due to uncertainties of its neutron emission and neutron spectrum. The computational analysis described in this article was performed in the preparation to the characterisation measurements.

1.2. Codes and tallies used

The calculation of the neutron production and transport in the NG was performed using MCNP 6.1 [2] or MCNPX 2.7 [3] and their derivatives, further described in section 2.1. Two types of tallies were used in the analysis: track length estimates of the neutron fluence averaged over the cell of interest (F4 tally in MCNP) and surface current tallies (F1 tally in MCNP). The track length estimate of the neutron fluence was used for the calculations of the angular dependence of the neutron fluence and spectrum in cells located on a thin spherical shell around the NG. Cells on a spherical shell were defined to describe regions 5° wide using conical surfaces (Figure 1). The surface current tally on a sphere that surrounds the NG was used for the normalization of results (explained further in 2.4).

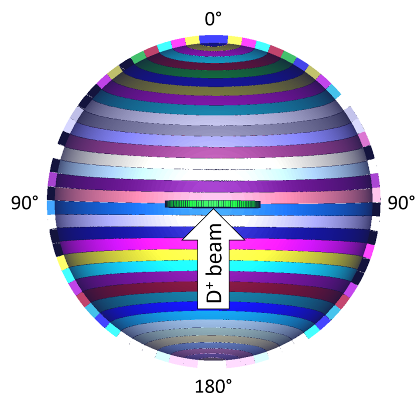

Figure 1: The cross-section of a spherical shell divided into cells describing 5° wide regions – the same as the tallies used in the analysis. The direction of the ion beam is denoted with the arrow while the target was located in the centre of the sphere. The neutron current tally was calculated on the inner surface of the spherical shell. The colours have no other significance than to help in distinguishing between different cells.

2. Simulation of a DT neutron source

2.1. Description of the codes

The calculation of the neutron emission from the DT neutron generator as a source of neutrons was performed using three different codes: the ENEA-JSI subroutine [4], MCUNED [5] and DDT. These three codes are based on independently developed models describing the DT neutron source and different additional capabilities.

The ENEA-JSI code is a subroutine of MCNP or MCNPX [3]. The simulation of the slowing down of D or T ions in the target material is based on the relatively old version (1996) of the SRIM code [6]. The evaluation of the scattering is performed using the analytic approximation called ”Magic Formula” [6]. The energies and angles of the neutrons produced via DT reactions in the target are calculated by application of two-body relativistic kinematics. Starting from the deuteron energy in the target frame, the neutron emission angle is sampled from the tabulated double differential cross section in the centre-of-mass frame at that deuteron energy. The resulting neutron emission angle and energy are then converted into the target frame. The deuteron deflections inside the target are taken into account by summing up the neutron emission angle in the target frame with the deuteron straggling angle. The transformation takes into account, usually negligible, relativistic effects. The ENEA-JSI subroutine was first developed to computationally support the analysis of the experiments at the Frascati Neutron generator (FNG) [7, 8, 9] and has been successively expanded to simulate a generic accelerator based DD/DT neutron generator. As the slowing-down of beam ions is modelled in the SRIM based code added to the MCNP, beam particles are tracked differently than other particles, e.g. neutrons and protons, in MCNP. As a result, the ENEA-JSI subroutine, can only be used in simulations of the (DD or DT) neutron source.

MCUNED is an extension of the MCNPX code. The standard MCNPX functionality is expanded so that the light ion nuclear data libraries in ACE format can be used in simulations of the ion transport through the material. This enables the calculation of the production of secondary particles through the interaction of ions with nuclei in the material based on the data from evaluated nuclear data libraries. The code is very general and can be used in many different applications [10] as the ions are treated the same way as all other particles in MCNPX. MCUNED has a built-in variance reduction method that forces the production of the secondary particles [5]. This can significantly speed-up the simulations where particles of interest are secondary particles produced via the interaction of ions with nuclei in the material.

DDT is a code based on the DRESS code [11] and produces the MCNP/MCNPX-readable source definition card (SDEF). The ion transport through the target of the NG is simulated outside the MCNP code using the SRIM code (TRIM module). Latest version, SRIM-2013, was used for the results presented in this paper. The information about the ion histories produced by SRIM is read by the DDT code where this information is used to produce the MCNP-compatible source definition card with appropriate angular dependencies of the intensity and energy distribution of emitted neutrons. The simulation of DT reaction includes relativistic effects. The advantage of the DDT code in our case is the fact that neutron source calculation is performed outside MCNP and is not repeated each time the neutron source is used in the MCNP calculation. Additionally, access to the MCNP/MCNPX source code is not required for the use of the DDT code and the DDT produced source definition card should work with all recent and future versions of MCNP/MCNPX.

2.2. Cross-sections and data

For the calculations of the neutron transport the neutron nuclear data used was mainly from FENDL-3.0 while the data from JEFF-3.2 was used for nuclides missing in FENDL. The only exception to this was the use of neutronic cross-sections from ENDF/B-VI.8 for the natural Ti in the target used for the calculations with the ENEA-JSI subroutine. The ion stopping power data from SRIM-2013 was used in the ENEA-JSI subroutine while the ion interaction cross-sections in MCUNED were from TENDL-2010 for titanium and ENDF/B-VI.0 for tritium. Nuclear data describing the DT fusion reaction was the same in all three codes – DT reaction cross-sections from ENDF/B-VII.1 used with the ENEA-JSI subroutine and DDT, and from ENDF/B-VII.0 used with MCUNED are identical.

2.3. MCNP model

A very simple MCNP model of the NG was used for the comparison of results produced by the three codes. The comparison focused on the DT reaction due to a D beam impinging on a tritiated target. As the purpose of this model was the comparison of codes, it included the necessary components only – the

Table 1: TiT target and ion beam parameters.

Parameter	Value
Target diameter	1.5 cm
Target thickness	2 μm
Target material	TiT_2
Target density	5.0 g/cm^2
D beam diameter	0.5 cm
D beam energy	100 keV

model consisted of a target containing titanium and tritium (TiT) mixture, and (empty) cells surrounding the target for tallying purposes. Additionally, the beam of D ions was defined to impinge on the target in the ENEA-JSI subroutine and MCUNED simulations to produce the neutrons via DT reaction in the target material. The source definition card from DDT was positioned very close to the TiT target (0.1 μm from the target material). The reason for this narrow gap between the source and the target is that in MCNP planar sources should never be defined on a surface defining a cell used in the geometry. The parameters of the ion beam and target were based on the information provided by the supplier and are presented in Table 1.

2.4. Renormalization of the results

The use of different codes that treat particles differently necessitates the renormalization of their results in order to make a comparison. The neutron tallies calculated by the ENEA-JSI subroutine and the DDT code are normalized to one emitted DT neutron while the results of MCUNED simulations are normalized to one ion in the ion beam. This means that of the three codes used in the comparison only the results of MCUNED contain the information about the number of neutrons produced per D ion in the ion beam – neutron yield. Due to the intended purpose of the comparison being the comparison of the three codes, a general renormalization procedure was developed to ensure that the same quantities were being compared. For this purpose, all results were divided by the renormalization constant suitable for each of the simulations. The chosen renormalization constant was the total number of neutrons emitted from the target per source particle (D ion) in the 4π solid angle which was calculated using the surface current tally on the sphere surrounding the target. For MCUNED simulations the renormalization constant is a value close to a neutron yield whereas the value for the other two codes is close to 1. The differences between the calculated values and these expected values are due to the neutron interaction with the target material (neutron absorption and multiplication). The results of calculations where the full model of the NG was used were renormalized using the normalization factors obtained for the case of the simplified model (TiT target) where the same ion beam and target parameters were used. This way all the results were renormalized to represent the same quantities – quantities normalized to one neutron emitted from the thin TiT target.

2.5. Code comparison

The angular dependence of the neutron fluence and the spectrum calculated with the three codes were compared. Results show that differences in physics models used in the simulations lead to differences in results. Both DDT and the ENEA-JSI subroutine use very similar models for the ion transport through the TiT target. In the former code the simulation is performed in the SRIM code while in the latter the simulation is performed in a code based on an older version of SRIM so the models used are, to a large extent, the same. Ion transport through the target and DT fusion reaction in MCUNED, on the other hand, are based on models from MCNPX.

A comparison of the angular dependence of the neutron fluences (Figure 2) and spectra (Figure 3), calculated as described in Section 2.1, show some agreements and some discrepancies. The angular dependence of the neutron emission and the energies of peaks in spectra at different angles are very similar

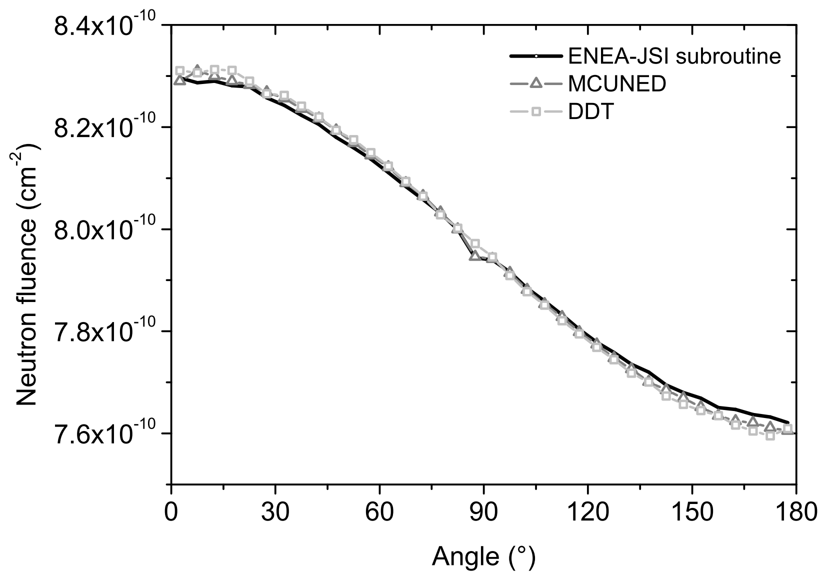

Figure 2: The angular dependence of the neutron fluence calculated using the three codes.

for all three codes. The spectra produced by the ENEA-JSI subroutine and DDT have similar shapes but differ in the widths of the peaks. Some significant differences between the results produced by MCUNED and the other two codes (DDT and ENEA-JSI subroutine) can be observed in the shape and width of the peaks in the directions that are not close to 0° or 180°. Our testing indicates that the difference most likely comes from the fact that the ion interaction in the target in MCUNED is based on the models from MCNPX while the ENEA-JSI subroutine and DDT use code based on SRIM or the SRIM code itself to model this interaction.

Because of the lack of experimental data where the results would determine, with high certainty, which of the three codes reproduces the experiments the most accurately, it was decided that the ENEA-JSI subroutine would be used as a reference source. This decision was based on the excellent reproduction of benchmark experiments performed at the Frascati Neutron Generator [12] using the ENEA-JSI subroutine. All calculations from here onward are thus

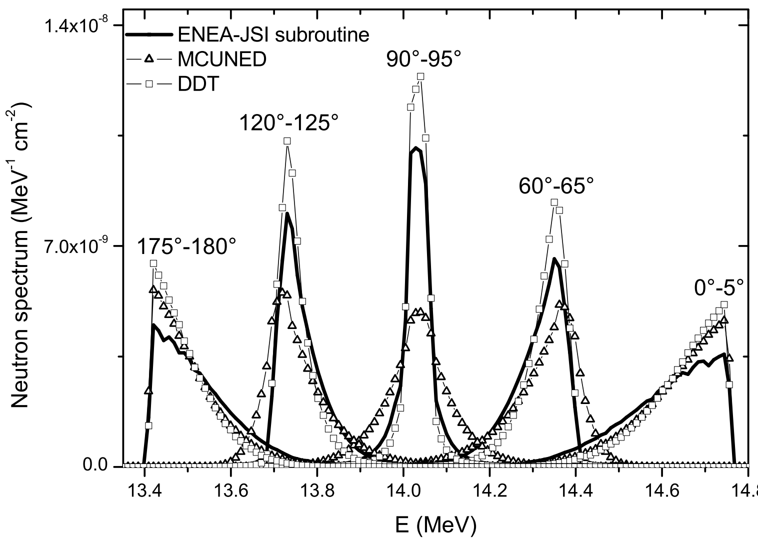

Figure 3: Neutron spectrum in five different directions calculated using the three codes.

performed using the ENEA-JSI subroutine unless stated otherwise.

3. MCNP model of the neutron generator

As the exact internal structure of the NG is proprietary, a sketch of the internal structure and material composition of the NG was provided by its supplier. This sketch was rather simplistic but was used as a reference for the modelling of the NG in MCNP because of the lack of more detailed information.

The lack of information on the detailed structure of the generator's interior introduces uncertainties to the simulations performed using this MCNP model. The effects of uncertainties in the geometry and material composition on the results of our simulations were assessed through sensitivity analyses where parameters with significant uncertainties were varied. To understand the effect of different parts of the model on the results of our simulations, the cells of the NG model were filled with their appropriate material in several stages (Figure

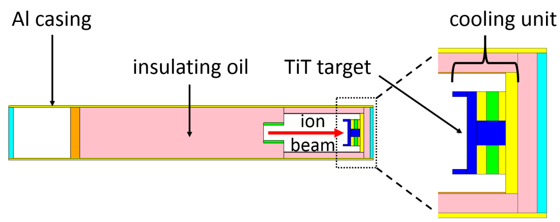

Figure 4: An MCNP model of the NG based on the sketch provided by the supplier. The colours denote the material in each cell. Most of the volume of the NG in this model is filled with the insulating oil (40% C, 60% H), there is copper (100% Cu) in the cooling unit (blue cells), two different aluminium alloys (cylindrical part of the Al casing: 94% Al, 6% Mg, front and rear surface of the Al casing and yellow part of the cooling unit: 94% Al, 2% Mg, 4% Cu), two kinds of steel (orange part: 70% Fe, 18% Cr, 10% Ni, 2% Mn) and the TiT target (described in Table 1).

5).

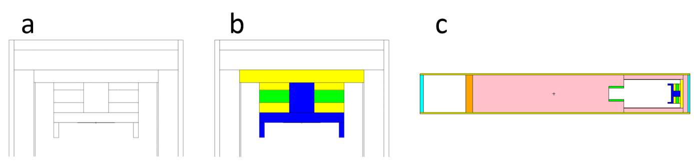

Figure 5: Multiple stages of the model assembly – TiT target only (a), TiT target and cooling unit (b), and full model (c) were filled with materials specified in a sketch by the NG supplier.

The largest effect of the NG materials was found to be on the angular dependence of the total neuton fluence (Figure 6) and on the amount of neutrons with energies below the energies of the DT peak. For example, the model in which only the thin TiT target was present produced 0.01% of neutrons with energies below 12.8 MeV, whereas the model of the NG where both the TiT target and the cooling unit were present produced 12% and full model 20% of these downscattered neutrons, respectively. As seen in Figure 6 the cooling unit causes a significant reduction of the neutron emission from the NG in the forward direction (angles between 0° and 90°) and increase in the backward direction (angles between 90° and 180°) while the rest of the NG's material significantly decreases the emission in the directions between 135° and 180° relative to the direction of

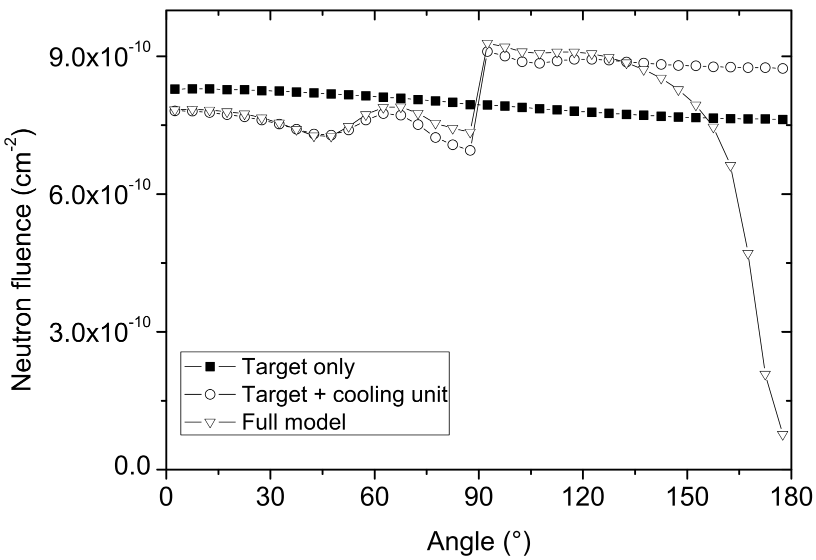

Figure 6: Neutron fluences calculated using models in multiple stages of the model assembly.

the ion beam.

A large part of the model is filled with a material corresponding to an insulating oil and the amount of metal objects in the model seems to be lower than we expected in such a generator (based on figures of similar neutron generators presented in [13]). To investigate the effects of metal objects on the neutron field, the insulating oil in the cells was replaced with mixtures of the insulating oil and stainless steel in various ratios. It was found that the effect of the replacement of the oil with stainless steel in most parts of the generator had minor effects on the neutron fluence and spectra. The difference was noticeable only when a significant amount of oil very close to the target was replaced with stainless steel, i.e. when oil in the cell surrounding the cooling unit was replaced with material where 90% of the mass was stainless steel. Even this drastic change in the material composition resulted in only a minor change of the neutron emission in some directions of the order of 2%.

If the ENEA-JSI subroutine or MCUNED are used for calculations where the NG is repeatedly used as the source of neutrons using the same NG settings, the calculation of the ion transport and neutron production are repeated each time the simulation is performed. In the case of DDT, or other source definition card based neutron sources, this is only performed once when the source definition card is produced. To decrease the required CPU time for calculations where the same NG is used multiple times, the neutron source characteristics, from the ENEA-JSI subroutine or MCUNED calculations, can be recorded and transformed into the source definition card description of the source (described in Section 3.1). To assess the relative efficiencies of the different codes, the ENEA-JSI subroutine, DDT and MCUNED, in the simulation of the DT neutron source, the figure of merit (FOM) values were compared to the FOM value of an isotropic 14 MeV point neutron source. FOM is defined as

$$FOM = \frac{1}{\sigma^2 \cdot \tau} \tag{1}$$

for one standard statistical uncertainty σ and CPU time τ and is, for a well converged result, independent of the number of simulated histories. CPU time used for the calculation of the FOM was the CPU time of the MCNP/MCNPX simulation only. The CPU time spent by the SRIM simulations, in the case of DDT, was not taken into account. The reason for this exclusion is that for our use the simulation of ion transport in the NG would only have to be performed once while MCNP simulations using the same neutron source will have to be performed for NG positioned on approximately 50 positions by the JET's remote handling system [14]. For cases where many different NG settings would be used, the CPU time spent by SRIM would have to be taken into account for a comparison to be relevant. FOM values were compared through the slow-down (SD) defined as

$$SD = \frac{FOM_{ref}}{FOM_{com}} \tag{2}$$

where FOM_{com} is the FOM of the simulation that is compared to FOM_{ref} the FOM of a refference simulation. SD is an indication of the relative amount of the

Table 2: Relative slow-downs of simulations compared to the reference simulation. ”Simple model” is the model used in the comparison of codes in section 2.3 while ”Full model” is a model based on the sketch from the NG supplier (described in section 3). VR indicates the variance reduction parameter in MCUNED that determines the number of neutrons that are forced to be produced per simulated ion history.

Neutron source, code	VR	Simple model	Full model
Point source, MCNPX	NA	1.00	3.58
Point source, MCNP 6.1	NA	1.46	4.92
DDT, MCNPX	NA	1.22	4.54
DDT, MCNP 6.1	NA	1.82	5.88
ENEA-JSI, MCNP 6.1	NA	6.61	10.9
MCUNED, MCNPX	1	34.6	147
MCUNED, MCNPX	10	4.48	19.7
MCUNED, MCNPX	100	1.33	6.25

CPU time needed to achieve the same statistical uncertainty as the reference simulation, e.g. $SD = 2$ means that the simulation requires two times more CPU time than the reference simulation. The reference simulation used in our comparisons was the simulation performed with MCNPX using a point isotropic 14 MeV neutron source in a simple model.

The results in Table 2 indicate that typically the MCNP simulation using the source produced by the DDT is the most efficient, followed by the ENEA-JSI source subroutine while the MCUNED simulations are somewhat slower. The ENEA-JSI subroutine and MCUNED calculations are slower than DDT due to the fact that in the former two codes the simulation of ion transport through the target is repeated each time the simulation is performed. However, the use of the variance reduction in MCUNED can significantly increase its efficiency even to a point where the CPU time of the MCUNED simulation is comparable to the simulation using the DDT defined source definition card.

3.1. Source definition card (SDEF) based neutron source

To experimentally calibrate the neutron detectors of the JET tokamak, measurements of the detector responses to a neutron source in various positions are performed. The computational support of the calibration experiment for a calibration to DT neutrons thus includes simulations of the detector responses to the NG in various positions. As the neutron source in all these calculations will be the same, the neutron source will be described using a standard source definition card (SDEF) to reduce the CPU time spent on the simulation of the neutron source. To produce the source definition card a simulation using the ENEA-JSI subroutine where the neutron emission and spectra were tallied at 100 angles was performed. The calculated tally values were then transformed into a source definition card using a modified ENEA–developed script.

A relatively accurate reproduction of the neutron source using the source definition card was demonstrated through comparisons of the simulations where the neutron source was reproduced using the source definition card and simulations using the ENEA-JSI subroutine.

The results in Figure 7 show that the spectra reproduced by the source definition card are very similar to the spectra of the original simulation (the simulation the source definition card is based on). The accuracy of the reproduction could further be increased by increasing the number of angles used in the reproduction.

Conclusions

A compact DT neutron generator will be used as a calibration source in the next calibration of the neutron monitors at the Joint European Torus. The computational support of the calibration experiment includes the modelling of the DT neutron generator used as a calibration source. An MCNP model of the neutron generator, based on the configuration and material information provided by the supplier, was constructed and tested. Three different codes capable of simulation of the DT fusion reaction in accelerator based systems (ENEA-

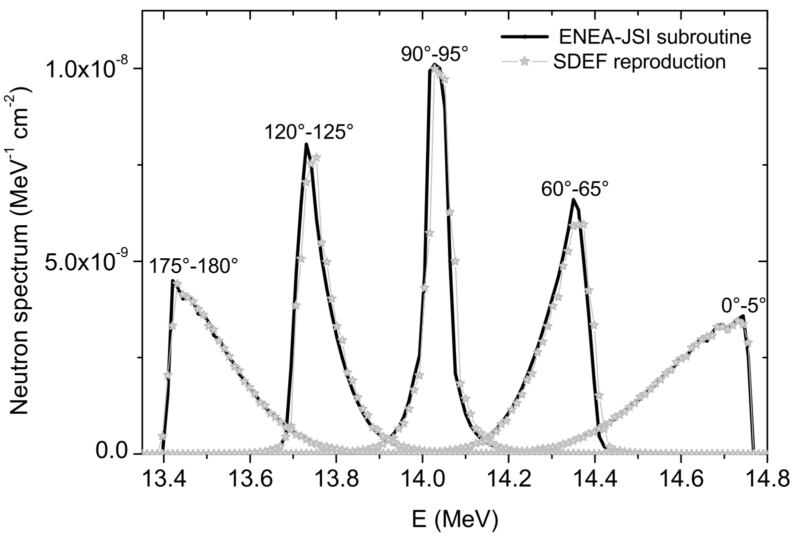

Figure 7: Neutron spectra in five directions for the simple model using the ENEA-JSI source subroutine and source definition card recording of the same source. The reproduction of the spectra with the source definition card is relatively accurate and could further be improved by increasing the number of angles where the spectrum is calculated for reproduction.

JSI source subroutine, MCUNED and DDT) were compared. The effects of materials in the models were calculated and sensitivities to uncertainties in the model analysed. The results of the simulations using this model showed a low sensitivity of results to uncertainties in the model composition. The model will be validated and improved based on the measured neutron fluences and spectra emitted by the neutron generator obtained in an experimental campaign [15]. The improved model will then be used in simulations replicating the calibration process of JET's neutron monitors. The results of the analysis described in this paper will contribute towards the 10% target accuracy of the calibration of the JET's neutron detectors and the 5% target accuracy of the neutron generator characterisation.

Acknowledgements

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

The work was supported by the Slovenian Ministry of Education, Science and Sport (project codes P2 0073 Reactor Physics, J2 6752 Development of methodology for calibration of neutron detectors with a 14.1 MeV neutron generator - JET fusion reactor case, J2 5467 Neutron calculations for use with neutron diagnostics - application to the JET fusion reactor).

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