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Application of the ADVANTG Hybrid Code on the JET3 – NEXP Streaming Benchmark Experiment

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An application of the ADVANTG Hybrid Code, which combines the deterministic transport solver Denovo with Monte Carlo code MCNP, on the JET3-NEXP streaming benchmark experiment is presented in this paper. An ADVANTG input parameter variation analysis was performed in order to find optimal input parameters for the hybrid two-step workflow. ADVANTG-accelerated calculations from three different institutions (JSI, ORNL and CCFE) were compared to analog MCNP simulations confirming no bias is introduced due to the use of ADVANTG. Additionally, ADVANTG accelerated MCNP numerical simulations of the neutron fluence were compared to experimental results performed in 2016 at JET using thermo-luminescence detectors (TLD). C/E values from 0.5 to 4.5 were calculated for the experimental positions in the SW labyrinth and SE chimney. Using ADVANTG-generated variance reduction parameters a speed-up of up to a factor of 1100 compared to analog calculations was achieved. ADVANTG has proven to be a powerful, user-friendly, and reliable tool for variance reduction of complex fusion streaming problems.

Keywords: ADVANTG, JET, NEXP, MCNP, variance reduction

1. INTRODUCTION

Due to the complexity of the geometry of realistic fusion devices, such as the ITER tokamak, and of the physical processes involved simulations of neutron streaming through ducts are extremely difficult. Experimental verification of the state-of-the art computer transport codes and nuclear data which will be used for these calculations is necessary. The JET tokamak offers a unique platform for testing transport codes and nuclear data because of its ITER like geometry and appropriate fusion neutron sources. The JET3-NEXP streaming benchmark experiment, currently underway at the JET tokamak, is a perfect platform for validating newly developed codes for neutron transport simulations and nuclear data used for simulations. It comprises of thermo-luminescence detector (TLD) measurements of neutron fluence at several positions around the JET tokamak torus hall which are compared to 3D Monte Carlo neutron transport simulations.

Monte Carlo simulations that do not utilize the variance reduction techniques or analog Monte Carlo simulations are in full-scale fusion problems computationally too expensive. Variance reduction is thus needed for comparison of the experimental results to the calculation. In a previous analysis of the streaming through the JET personnel labyrinth various variance reduction techniques were used such as the MCNP weight-window generator, the in-house CCFE WWITER variance reduction generator [1] and the surface source write (SSW)/read (SSR) options available in MCNP [2]. These methods proved to be successful but time-consuming both in the sense of the time needed for the user to create effective variance reduction parameters and in the sense of the actual simulation times.

Recently a new hybrid deterministic/Monte Carlo code, ADVANTG [3], [4], [5], [6] and [7] developed by the Oak Ridge National Laboratory (ORNL), was released. ADVANTG's approach is to use the deterministic discrete ordinates transport code Denovo [8] to calculate forward and adjoint fluxes that are used to generate variance reduction parameters specifically weight windows (WW) and source biasing parameters.

The second part of the paper describes the two computer codes used to perform the calculations of the neutron fluences at the experimental positions. The third part of the paper briefly describes the JET3-NEXP experiment including the compositions and locations of the TLDs.

In the fourth part of the paper results of an ADVANTG input parameter variation analysis are reported. Optimal parameters were chosen to determine variance reduction parameters to efficiently calculate neutron fluences at all experimental positions throughout the JET tokamak building.

The results of the hybrid two-step ADVANTG/MCNP workflow are presented in the fifth part of the paper. These results are compared to analog calculations to confirm no bias was introduced to the results using ADVANTG generated variance reduction parameters. A comparison between the calculation results from three different institutions Jožef Stefan Institute (JSI), Oak Ridge National Laboratory (ORNL) and the Culham Centre for Fusion Energy (CCFE) is also presented in the fifth part of the paper. Finally, the calculation results are compared to experimental results.

2. DESCRIPTION OF COMPUTATIONAL METHODS

The hybrid deterministic/Monte Carlo code ADVANTG utilizes the Denovo S_n discrete ordinates deterministic solver to determine variance reduction parameters in a format compatible with the Monte Carlo transport code MCNP. This section briefly describes the ADVANTG and MCNP codes.

2.1. ADVANTG

ADVANTG, an **A**utomate**D** **VA**ria**N**ce reduc**T**ion **G**enerator [3], automatically generates variance reduction parameters for neutron and photon transport problems defined in an MCNP input file. The code package was developed by ORNL. The main purpose of the code package is to reduce the time needed by the user to generate appropriate variance reduction parameters and to accelerate Monte Carlo simulations in terms of the CPU time. Space- and energy-dependent mesh-based WW, and a biased source distribution are generated by ADVANTG based on a three-dimensional discrete ordinates solution of the adjoint and forward transport equation. The relation between the forward and adjoint transport operator is given by equation (1) as

$$\langle \psi^{\dagger}, H\psi \rangle = \langle \psi, H^{\dagger}\psi^{\dagger} \rangle, \tag{0}$$

where ψ is the angular flux, H the transport operator, and $\psi^{\dagger}$and$H^{\dagger}$ their adjoint counterparts. It is well known that the adjoint flux is a measure of the importance of a neutron contributing to the response of an arbitrary detector [9]. To generate variance reduction parameters, more specifically WW parameters, ADVANTG uses this fact in the form of the Consistent Adjoint Driven Importance Sampling CADIS [1] method, where the statistical weight of particles$w$ is proportional to the inverse of the adjoint flux. The relation is given with equation (2) as

$$w(P) = \frac{R}{\psi^{\dagger}(P)}. \tag{0}$$

where R is referred to as a response or quantity of interest and $P = (\acute{r}, E, \hat{\Omega})$ the phase-space of position vector, energy, and solid angle. One should be aware that ADVANTG uses scalar (directionally integrated) fluxes in order to determine the variance reduction parameters and not angular fluxes as is the case in the general form of the CADIS methodology given in equation (2). The required forward and adjoint deterministic transport calculations are performed by the Denovo [8] S_n discrete ordinates deterministic transport solver, which is part of the ADVANTG code package.

The Forward-Weighted CADIS (FW-CADIS) [10] method is used when variance reduction parameters suitable for global transport calculation or for simultaneous speed-up of calculations of multiple tallies of interest are needed. A detailed description of the two methods are given in the references [1], [3], [4] and [10].

ADVANTG uses ray-tracing techniques to automatically convert the MCNP geometry model and material information defined in an MCNP input file to a deterministic Cartesian model suitable for Denovo transport simulations.

A WW file in a format compatible with MCNP and a modified MCNP input file with source biasing parameters and input cards needed to read the separate WW file are produced by ADVANTG which are both needed for the consequent MCNP calculation. Additionally, the results of the deterministic calculation, for example the total forward and adjoint fluxes by energy group, Cartesian model informant, deterministic mesh, and deterministic source distribution are written to Silo-format files. The files can be visualized using the VisIt visualization tool available from the Lawrence Livermore National Laboratory or the VisIt website [11].

2.2.MCNP

MCNP, a General **M**onte **C**arlo **N**-**P**article Transport Code, developed by the Los Alamos National Laboratory [12], is a general-purpose neutron/photon/electron Monte Carlo transport code. Individual particle events are stochastically sampled in order to simulate particle transport. Individual particle histories are tallied when in regions of interest to estimate their average behavior. MCNP is widely used in the particle transport community all around the world. It has been applied and validated on several benchmark experiments [13]. The problem geometry, material, cross-section, source, and tally of interest information are all defined in a user-generated input file. The input file used in this paper was provided by the JET community and only the tally and neutron cross-section parts of the input were modified. MCNP5 program code version 1.6 was used for all simulations performed in this work.

3. DESCRIPTION OF THE EXPERIMENT

In 2012 the first batch of several hundreds of very sensitive thermo-luminescence detectors enriched to different levels of ^{6}LiF/^{7}LiF were positioned at several experimental positions all over the JET tokamak torus hall. The experimental campaign from 2012 was repeated during the 2013 – 2014 JET campaign and the 2016 JET Deuterium – Deuterium (DD) campaign. Lessons learned in the previous campaigns were used to improve the experimental configurations. Highly sensitive ^{nat}LiF: Mg,Cu,P (MCP-N) and ^{7}LiF:Mg,Cu,P (MCP-7) detectors were used. The MCP-N TLDs contained natural lithium with 7.59 % of ^{7}Li and the MCP-7 TLDs were enriched to 99.97 % ^{7}Li. Only these two kinds of TLDs were used in the 2016 campaign in order to avoid interference between TLDs due to self-shielding effects. The two different enrichments of lithium are used to differentiate between neutron and non-neutron components of the radiation field. The capture cross-section for neutrons of ^{7}Li is significantly lower compared to the capture cross section of ^{6}Li and thus, TLDs with low concentration of ^{6}Li are almost insensitive to neutrons.

Two high-density polyethylene containers for the TLDs, one in vertical orientation (square shape) and the other in horizontal orientation (circular shape), were used to investigate the shadowing effect in the directional neutron filed. Natural lithium (N) and enriched ^{7}Li (7) TLDs were evenly arranged in the two polyethylene containers. Both of the containers were immersed in a large high density polyethylene moderator. The configurations of the TLDs and the polyethylene containers are shown in Figure 1. The TLDs were designed, produced, calibrated, and analyzed at the Institute of Nuclear Physics (IFJ) in Kraków, Poland. More detailed information on the TLDs, their composition and calibration techniques are reported in the provided references [1], [14] and [15].

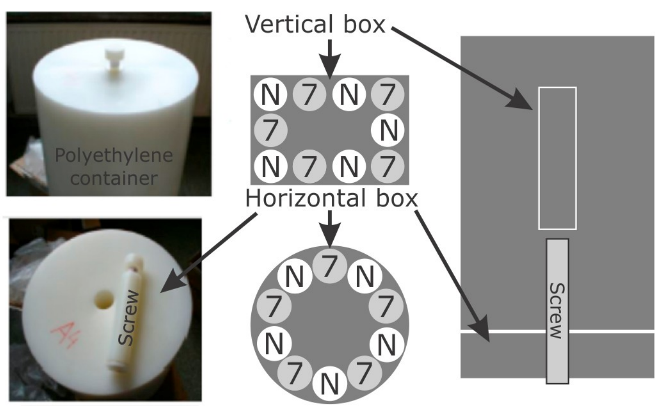

Figure 1: TLD assemblies of N: MCP-N and 7: MCP-7 TLDs in high density polyethylene holders and container.

The TLD assemblies were positioned in 16 different locations (A1 – A8, B1 – B8[1]) at different distances from the plasma neutron source in the tokamak. The furthest positions in the South West (SW) labyrinth and in the Torus Hall basement through the air duct chimney are about 40 m from the plasma neutron source. The SW labyrinth is a five segment doglegged corridor with a side wall made of borated concrete. Three large air ducts – the chimney – connect the Torus Hall to the basement through the Torus Hall floor in the South East (SE) corner. In the basement, several streaming paths originate at the bottom

[1] Results for location B8 were omitted from this paper due to problems with the measurement.

of the chimney. The TLD assembly locations are shown in Figure 2. Note that only the x and y coordinates are represented accurately in Figure 2 and the z coordinates differ and are given in Table 2.

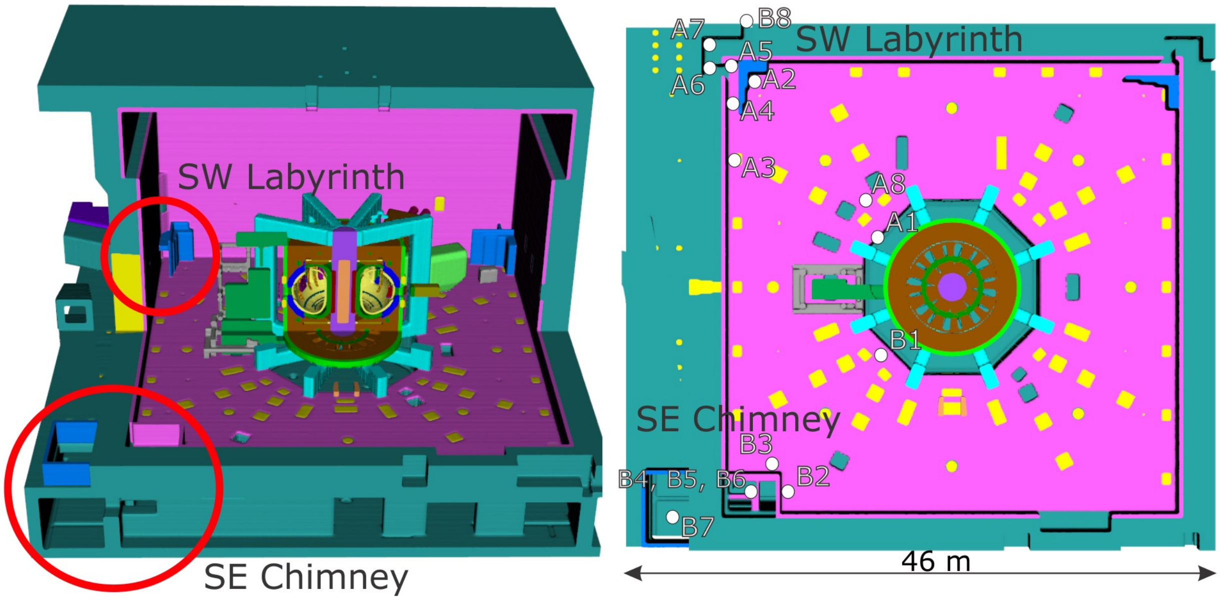

Figure 2: Experimental locations of the TLD assemblies throughout the JET tokamak torus hall. Note that only x and y coordinates are representative but the height coordinate varies. See Table 2 for details on the locations[2].

4. CALCULATIONS

A two-step workflow was implemented for the numerical simulations of the neutron flux at all experimental positions. In the first step, ADVANTG is used to produce variance reduction parameters for the second step MCNP calculation. The calculations in this paper were performed using MCNP5 version 1.60 and ADVANTG version 3.0.1. The computation nodes used in the analysis use 2 Intel Xeon E5-2680 v2 processors with 2×10 (2.8 GHz) cores resulting in 40 processor threads, and 128 GB of DDR3 memory. Additionally a computer node with 2 Intel Xeon E5-2697 v3 with 2 × 14 (2.6 GHz) cores and 256 GB of DDR3 memory was used for the ADVANTG calculations with the largest number of voxels in the geometrical mesh and the calculations with the 200n47g and FENDL253 multigroup data libraries because of the larger computer memory requirements.

In this part of the paper, some relevant details on the full 360⁰ MCNP model of the JET tokamak are given. Conclusions on an extensive ADVANTG input parameter variation are also presented including the effect of the multigroup data library variations, quadrature set variations and the effect of the choice of the geometrical mesh for the deterministic calculation.

4.1. MCNP model information

Because of additional difficulties with the simulations inside of high-density – high-attenuation TLD assemblies, attenuation factors were defined in order to replace the actual polyethylene TLD containers in the MCNP model. Attenuation factors calculated with a model of the TLD assemblies and a plane source

[2] Note: the MCNP model is rotated 90⁰ clockwise compared to the blueprints.

with an appropriate neutron energy spectra were calculated by CCFE using MCNP. The energy-dependent tally modification attenuation factors are given in Table 3. The TLD attenuation factors are an average of the response for the circular and square detectors and are independent on the detector positon in the tokamak torus hall. An analysis of the effect of the attenuation factors is planned. These factors are part of the MCNP input file so no further calculations with them are needed.

The MCNP integrated DD fusion neutron source spectra was used as the energy distribution of the source. The shape of the plasma source corresponds to the typical shape of the plasma in the JET tokamak [16].

The default nuclear data libraries for all JET simulations are the FENDL libraries, more specifically the latest FENDL 3.1b library [17] was used. In order to fulfil the requirement of using this library a complete material redefinition of the MCNP input was performed. A redefinition of the materials to only contain per-isotope entries was performed using the MATSSF [18] code. Some isotopes contained in the MCNP input are not included in the FENDL 3.1b library. The JEFF 3.2 [19] library was used for the following isotopes missing from FENDL: ^{75}As, ^{84}Sr, ^{86}Sr, ^{87}Sr, ^{88}Sr, ^{113}In, ^{115}In, ^{144}Sm, ^{147}Sm, ^{148}Sm, ^{149}Sm, ^{150}Sm, ^{152}Sm, ^{154}Sm, ^{151}Eu, ^{153}Eu, and ^{234}U. Since there are only trace amounts of these isotopes they do not affect the final results in a significant way,

Different tallies were defined in the MCNP model. JSI and CCFE used the track length estimator (F4 tally in MCNP nomenclature) tallies averaged to the volume of the actual TLDs. ORNL used point detector tallies (F5 tally in MCNP nomenclature) positioned in the center of the TLD assemblies with an exclusion radius of 10 cm.

4.2. Effect of ADVANTG input parameter variations

The choice of the ADVANTG input parameters can significantly affect the CPU time, memory requirements, converging of the results, computational bias, etc. By choosing the appropriate Denovo solver, multigroup cross section libraries, S_n angular approximation, P_n order, and mesh size/density the user can optimize the final runtime and convergence of the accelerated Monte Carlo simulation.

An extensive ADVANTG input parameter variation analysis was performed on a simplified model of a JET-like tokamak in 2016 [6]. The lessons learned were applied to the full-size detailed JET tokamak model [20]. This time, only the input parameters which were found to have a significant impact on the convergence of the results were varied to find the final optimal variance reduction parameters used to calculate the results reported in this paper. Three of the most important parameters were the choice of the multigroup data libraries used for the deterministic Denovo calculation, the choice of the angular quadrature sets, and finally the definition of the deterministic mesh. In this article section the effects of these three parameter variations are reported. In order to identify optimal ADVANTG input parameters, MCNP statistical tests with an emphasis on the statistical quantity Figure-of-merit (FOM) for location A7 in the SW personnel entrance labyrinth were compared. MCNP performs statistical tests on the calculated tally estimates in order to test the mathematical reliability of the convergence. One of the quantities used in the statistical tests performed by MCNP is the FOM defined with equation (3) as

$$FOM = \frac{1}{T * R^2}, \qquad (0)$$

where T is the CPU time of the simulation and R the relative statistical uncertainty of the tally. It can be interpreted as the efficiency factor in terms of the CPU time needed where higher FOM value is better. For a well-sampled problem, the FOM value is independent of the number of simulated particle histories.

Only the MCNP computational time was taken into account in the *T* variable as the ADVANTG calculation took a maximum of 2 hours with the densest geometrical mesh and the DPLUS multigroup data library which is roughly 8 % of the total two-step workflow.

Results of the parameter variation analysis are compared to calculations performed with *refined* parameters which were refined over several iterations of the two-step workflow. These parameters include the use of the DPLUS multigroup data nuclear data library, a refined geometrical mesh, and a detailed quadrature set. With these parameters, the default neutron fluence $(\Phi$¿¿$Refined)$¿ and FOM $(FOM$¿¿$Refined)$¿ were calculated. In subsections 4.2.1., 4.2.2., and 4.2.3. one of the three most important parameters (multigroup data library, angular quadrature set, deterministic geometrical mesh) is varied and the others are kept at *refined* values.

4.2.1. Multigroup data libraries

In deterministic calculations of neutron transport with codes such as Denovo, the choice of nuclear data library is very important not only because of the underlying source data (ENDF/B-VII.0 [21], FENDL-3.1 [16]) but also because of the energy group structure and the weighting functions used to collapse the data.

ADVANTG or rather Denovo uses ANISN-format nuclear data libraries for particle transport simulations. The libraries included with the ADVANTG code package are based on various nuclear library evaluations and were obtained using different weighting functions as they were intended for different applications. Of primary interest were the nuclear data libraries intended fusion related problems and general shielding problems based on the ENDF/B-VII.0 (27n19g, 200n47g, DPLUS) and FENDL-3.1 (FENDL71 and FENDL253) nuclear data evaluations.

The 27n19g and 200n47g libraries are general-purpose shielding libraries, which consist of 27-neutron/19-gamma groups and 200-neutron/47-gamma groups respectively. These libraries were obtained using a fission spectrum, a 1/E slowing down part of the spectrum and a Maxwellian distribution as weighting functions. The FENDL71 and FENDL253 libraries consist of 47-neutron/24-gamma groups and 211-neutron/ 42-gamma groups. They were produced by the ADVANTG team and are based on the FENDL 3.1b library. Only preliminary testing of the FENDL based libraries has been performed by ORNL and by the JSI team as they have not been publicly released with ADVANTG. The DPLUS library was processed from the ENDF/B-VII.0 nuclear data library using weighting functions based on the DABL-69 [22] library by the ADVANTG team. DABL69 is a 46-neutron/23-gamma group library that was developed for defense-related radiation shielding applications and consists of the weighting function similar to the one from the 200n47g library with an added 14MeV DT fusion neutron peak.

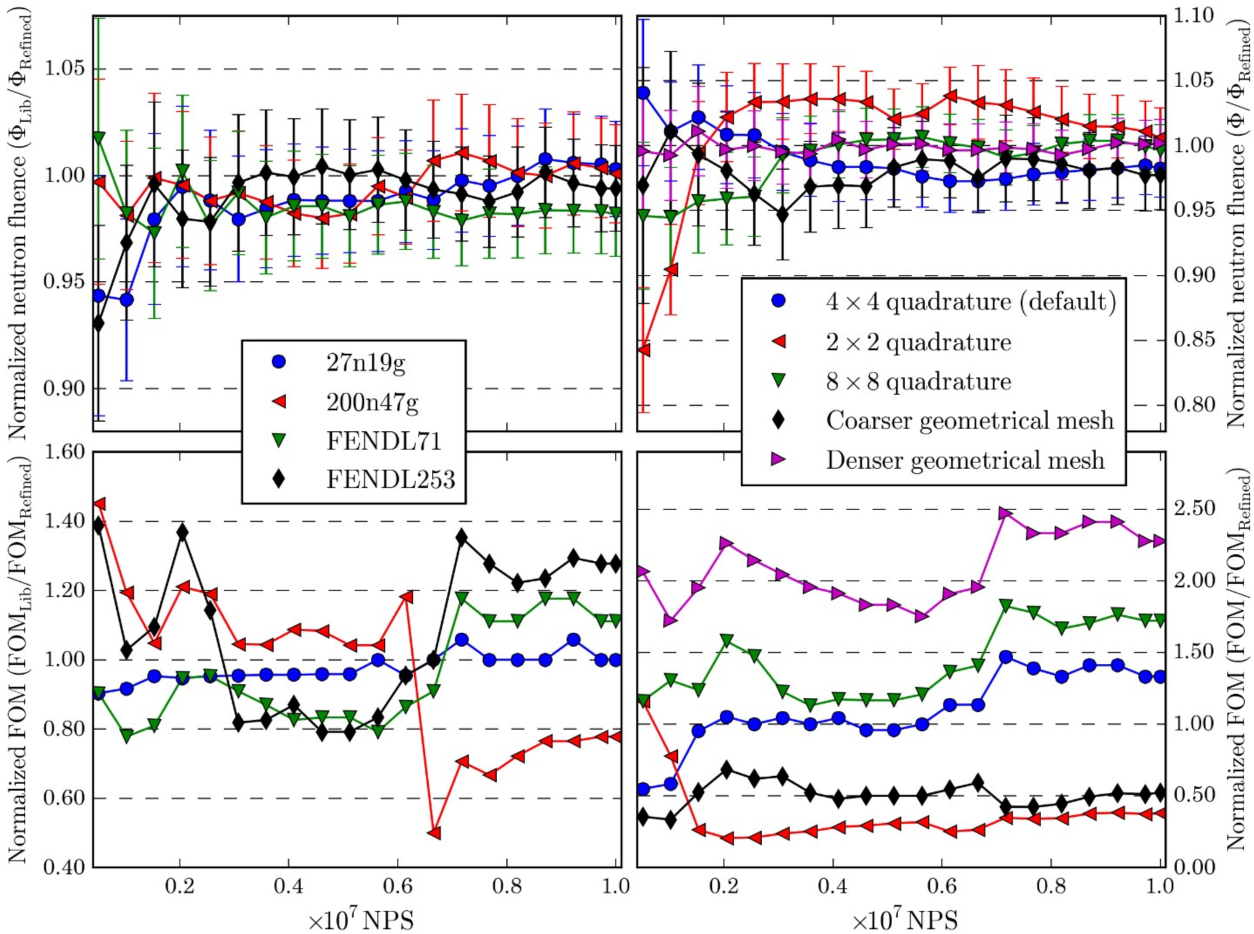

Figure 3: Normalized value of neutron fluence (top) and normalized FOM (bottom) calculated with 4 different multigroup data libraries (left), 3 different quadrature sets (default, 2×2 and 8×8) and 2 different geometrical meshes (coarser and denser) (right). The values as a function of NPS are normalized to the calculation with the *refined* parameters$(\Phi ¿¿ Refined)¿$. All of the results are calculated with MCNP and ADVANTG generated weight windows with different ADVANTG input parametrs. The error bars represent the combined uncertainty of the ratio.[3]

In the top-left subfigure of Figure 3, the neutron fluence at TLD location A7 calculated with MCNP and ADVANTG generated weight windows based on the Denovo deterministic calculation with 4 different multigroup data libraries $(\Phi ¿¿ Lib)¿$ normalized to the calculation with the *refined* input parameters is shown as a function of the number of simulated particle histories (NPS). The error bars were calculated by adding the errors of simulation with *refined* parameters and the simulation with different multigroup data libraries according to equation (4). The top-left subfigure of Figure 3 shows that even when using different libraries the mean value converges to the same value within the statistical uncertainty after sufficient particle histories have been simulated.

In the bottom-left subfigure of Figure 3 the FOM statistical tests normalized to the calculation with *refined* input parameters for TLD location A7 is shown as a function of the number of simulated particle histories (NPS) for 4 different multi-group data libraries. All of the multigroup data libraries performed

[3] Calculations were performed only at NPS values designated with symbols. Lines connecting the “points” act as eye-guides only.

similarly but the FENDL71 and FENDL253 libraries performed slightly better which was expected because of the fusion dedicated group structure and underlying FENDL 3.1b cross sections.

4.2.2. Quadrature sets

The number of angles to be used in the angular discretization of the deterministic Denovo calculation and different types of the angular quadrature sets can be chosen by the user. The choice of angular quadrature sets can significantly affect the final deterministic results and consequently the variance reduction parameters. The default type, the quadruple range (QR) quadrature set, was used in this analysis with a varying number of azimuthal and polar angles.

In regions with low scattering materials the so-called “ray effects” can be observed if a low number of azimuthal and polar angles are chosen. Ray effects are features of S_n deterministic transport solutions where the solution is unphysically more pronounced in certain directions seen as rays of high particle flux in the directions defined by the quadrature set angles. In the case of the JET streaming benchmark ray effects were observed when using the default quadrature set settings (4 azimuthal and 4 polar angles). Thus a detailed quadrature set was defined for the need of this analysis with 8 polar angles and the number of azimuthal angels was set to monotonically increase by 1 azimuthal angle per polar angle from 1 at the vertical polar angle to 8 at the horizontal polar angle (equator) – forward peaking quadrature set. This quadrature set was chosen as the *refined* parameter for all calculations but a variation of the number of angles was performed and the effect on the final statistical tests was studied. Besides the *refined* quadrature set the default quadrature set (4 azimuthal and 4 polar angles - default) was used and a quadrature set with 2 times more (8 azimuthal and 8 polar - 8×8) and 2 times less (2 azimuthal and 2 polar - 2×2) angles.

In the top-right subfigure of Figure 3, the neutron fluence calculated with 3 different quadrature sets (default, 2×2 and 8×8) (Φ) normalized to the calculation with the *refined* input parameters at TLD location A7 is shown as a function of the number of simulated particle histories (NPS). The error bars were calculated by adding the errors of simulation with *refined* parameters and the simulation with the different quadrature sets according to equation (4). The mean values of the three calculations are equal to 1 within the statistical uncertainty after the simulations have converged.

In the bottom-right subfigure of Figure 3, the FOM statistical tests for TLD location A7 are shown as a function of the number of simulated particle histories (NPS) for 3 quadrature sets. As expected the quadrature set with the highest number of angles (8×8) performed best, but at the cost of extra computer memory needed for the calculation.

4.2.3. Geometrical mesh density

The most time consuming and the crucial part of preparing ADVANTG input file is the definition of geometrical mesh. Denovo only supports Cartesian geometry. Several aspects of the neutron transport problem have to be taken into account such as the mean free path of neutrons in materials of interest and the dimensions of objects describing the geometry. In the case of the JET streaming benchmark, several major objects with varying sizes have to be taken into account such as the ports of the vacuum vessel and the details of the SW labyrinth and SE chimney. Because of the computer memory limitations in depth knowledge on the major neutron streaming paths is necessary to define an appropriate mesh. Several iterations were performed in order to refine the mesh for each of the measurement locations in the JET streaming benchmark. Appropriate meshes were identified by checking the direct and adjoint flux solutions alongside the contributon fields. When the deterministic solution yielded physically appropriate and expected values, a MCNP calculation was performed with the resulting variance reduction parameters.

If the statistical tests of the MCNP calculation were passed the mesh was deemed suitable and if any of the tests failed the mesh was refined further. The number of voxels for each of the measurement locations varied from about 900 thousand to 10 million.

Besides the *refined* mesh two additional meshes were used. A denser mesh with 2-times more voxels and a coarser mesh with 2-times fewer voxels. A comparison of all mesh iterations would not reveal information physically relevant to this paper. In the top-right subfigure of Figure 3, the neutron fluence calculated with 2 additional geometrical meshes (coarser and denser) (Φ) normalized to the calculation with the *refined* input parameters at TLD location A7 is shown as a function of the number of simulated particle histories (NPS). The error bars were calculated by adding the errors of simulation with *refined* parameters and the simulation with the different geometrical meshes according to equation (4). The mean values of the two calculations are equal to 1 within the statistical uncertainty after the simulations have converged.

In the bottom-right subfigure of Figure 3, the FOM statistical tests for TLD location A7 are shown as a function of the number of simulated particle histories (NPS) for the two different meshes. An increase of the relative FOM of about a factor of 2 is seen when using the denser mesh but once again at a large cost of the extra computer memory needed for the calculation. A factor 2 decrease in the relative FOM is seen when using a coarser mesh. This is probably because of the homogenization of materials in major neutron streaming paths which can cause particle over-splitting and in turn longer calculation times and lower FOMs.

5. RESULTS

5.1. Comparison against analog simulations

To ensure that ADVANTG does not introduce bias into the ADVANTG accelerated (AA) MCNP simulations a comparison to analog MCNP simulations was needed. As stated in the introduction analog simulations of this complex streaming problem are extremely time-consuming. In order to get some statically relevant results from the analog simulation in a reasonable time, point detector tallies (F5 tallies in MCNP nomenclature) were used and a simulation with $5 \cdot 10^8$ particles was performed. The total simulation time was roughly 13 days (wall-time) on 18 cores on the computer cluster mentioned in section 4. This analog simulation produced results with a statistical uncertainty below 5 % for locations A1, A8, A2, A3, A4, B1, B2, B3, B4, B5 and B6. The most challenging locations for the simulation A5, A6, A7 and B7 still had statistical uncertainties above 10 %. To further ensure that ADVANTG does not introduce bias in to the simulations the calculations were performed by three different institutions: JSI, ORNL and CCFE. All of the institutions used the same original MCNP input files but produced their own ADVANTG input files and consequently different variance reduction parameters.

In the top left subfigure of Figure 4 ADVANTG accelerated (AA) MCNP simulations of neutron fluences from three different institutions JSI, ORNL and CCFE are compared to the analog simulations for all TLD locations. In the top right subfigure of Figure 4 the ADVANTG accelerated MCNP calculations are normalized to the analog values ($\Phi_{norm} = \Phi_{AA} / \Phi_{Analog}$). The error bars were calculated by adding the errors of the ADVANTG accelerated (AA) simulations and the analog simulations according to equation (4)

$$\sigma_{norm} = \Phi_{norm} \sqrt{\left(\frac{\sigma_{AA}}{\Phi_{AA}}\right)^2 + \left(\frac{\sigma_{Analog}}{\Phi_{Analog}}\right)^2} \qquad (0)$$

One can observe that for locations with a low statistical uncertainty of the results i.e. A1, A8, A2, A3, A4, B1, B2, B3, B4, B5 and B6 the ratio Φ_{norm} is mostly 1 within 1 σ statistical uncertainty. Because of the large statistical uncertainty of the results at the remaining experimental locations the point detector results are untrustworthy [12] and should be disregarded. A longer analog simulation should be performed to ensure that no bias was introduced even for this locations but this was computationally too expensive for the scope of this work. But it must be said that despite the large statistical uncertainty of the analog results for locations A5, A6, A7 and B7 the Φ_{norm}ratios remained close to 1.

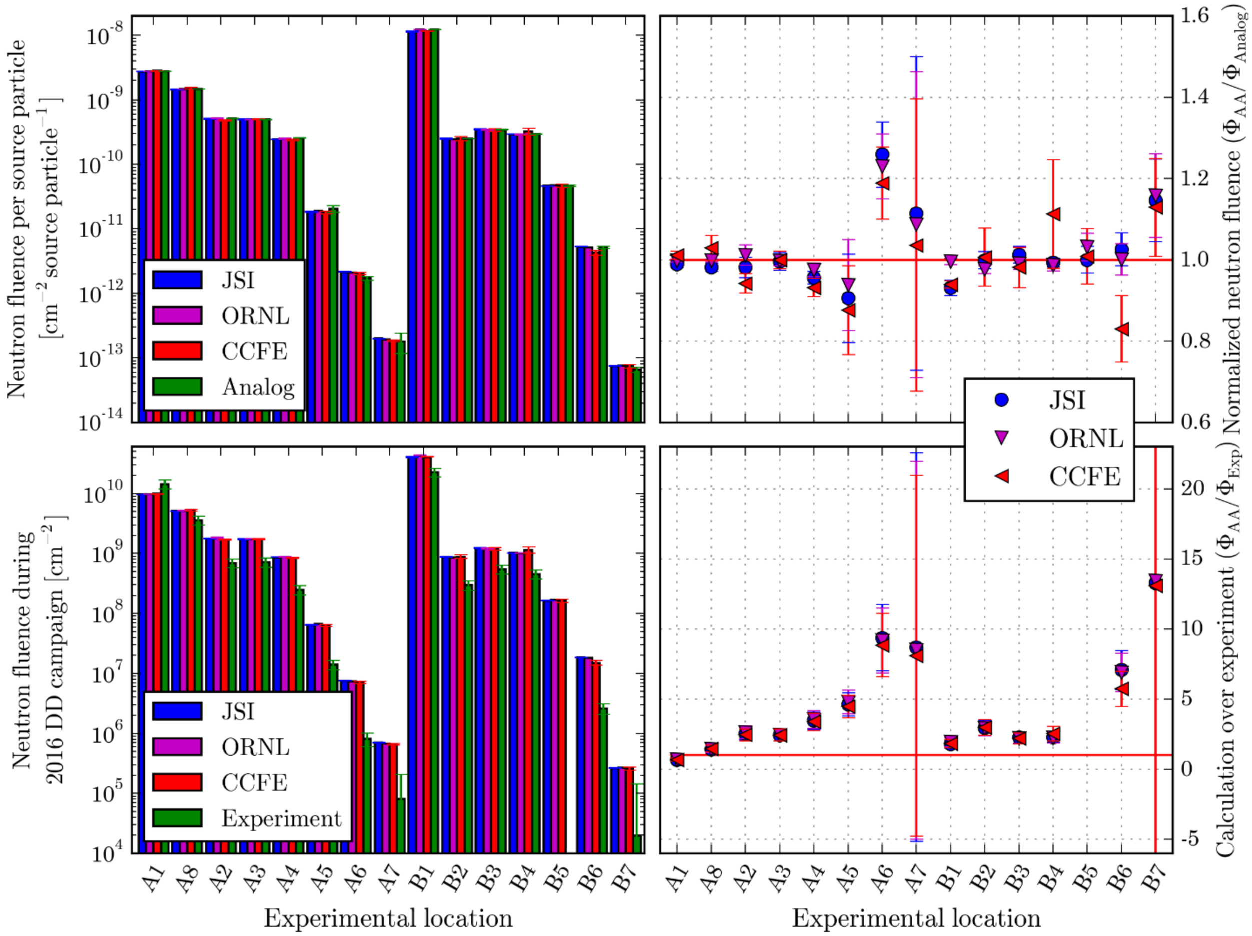

Figure 4: A comparison of absolute (left) and relative (right) values of the calculated neutron fluence for all TLD locations determined by three different institutions: JSI, ORNL and CCFE. The ADVANTG accelerated neutron fluence calculation (Φ_{AA}) is compared and normalized to the analog calculation (top) and to the experiment (bottom). The error bars represent the combined uncertainty of the ratio[4] (top right) or the combined uncertainty of simulations and experimental values (bottom right).

5.2. Comparison against the experimental data

In the previous subsection, we compared ADVANTG accelerated simulations to analog MCNP simulations. Overall, no significant bias was observed for locations with statically significant results. With such a comparison confidence in AA simulations was gained and the results could be compared to the experimental results which is one of the tasks of the JET3-NEXP streaming benchmark experiment. The basics of the experiment are described in section 3 and additional information can be found in the

[4] The relative uncertainty of the Calculation over Experiment value for location B7 is 630 %.

provided references. The final results were normalized to the total neutron yield of the 2016 JET Deuterium – Deuterium (DD) campaign of $3.518 \cdot 10^{18}$ neutrons. An average of the results from the circle and square TLD orientations is taken as the experimental result. The uncertainty of the TLD measurements is below 5 %. The complete experimental uncertainty including the uncertainty of the calibration of TLDs, the actual TLD measurements, neutron yield and the background radiation measurement is below 25 % for locations A1 – A6 and B1 – B6. The farthest locations from the plasma A7 and B7 have experimental uncertainties of 150 % and 630 % respectively. This is due to the fact that both measurements are almost at the level of the background measurement and because of the large uncertainty of the background radiation which is 240 %.

In the bottom left subfigure of Figure 4 ADVANTG accelerated (AA) MCNP simulations of neutron fluences from three different institutions JSI, ORNL and CCFE are compared to the experimental results (Exp) for all TLD locations. In the bottom right subfigure of Figure 4, the ADVANTG accelerated MCNP calculations are normalized to the experimental values ($C/E = \Phi_{AA}/\Phi_{\mathrm{exp}}$). The combined AA and experimental uncertainty error bars were calculated analogous to the ones in the previous section.

A trend in the C/E values can be observed. Generally, the calculations further away from the plasma in the SW labyrinth and SE chimney over predict the neutron fluence compared to the experimental results. There are several reasons for this discrepancy from the standpoint of the simulations such as modelling simplifications in the MCNP model, the lack of actual TLD polyethylene containers in the MCNP model and connected with that the directional component of the neutrons close to the plasma source. Additionally, the uncertainties of the calculations which arise from the uncertainties in the actual nuclear data cross sections have not been quantified yet. These uncertainties are expected to be large due to the large number of interactions that the neutrons undergo before they reach the experimental locations evident from the drop in neutron fluence of several orders of magnitude.

5.3. ADVANTG acceleration of simulations

The main purpose of the variance reduction assisted Monte Carlo calculations is to obtain unbiased detector response results with a sufficiently low statistical uncertainty in as low CPU time as possible. ADVANTG achieves these by generating WW and source biasing parameters on the basis of a fast global deterministic calculations. In order to compare the efficiency of the analog and AA calculation in terms of the required CPU times, we can compare the Figure-of-merit statistical quantity. The test is described in subsection 4.2. The speed-up factor or relative figure-of-merit ($\mathrm{FOM_{rel}}$) is defined as the following ratio

$$FOM_{rel} = \frac{FOM_{AA}}{FOM_{Analog}}, \tag{0}$$

where FOM_{AA} and FOM_{Analog} are FOM values for a ADVANTG accelerated (AA) and an analog MCNP simulation respectively.

In Table 1 the final FOMs are listed for both F4 and F5 tallies. Both analog and AA calculations were performed on the same computer cluster with the same number of cores and the same number of simulated particle histories. Both of the tally FOMs are given because in the analog simulation no F4 tally scores occurred. In the last two columns, the $\mathrm{FOM_{rel}}$ factors are calculated for each tally location. The $\mathrm{FOM_{rel}}$ factors for locations in the SW labyrinth (A1 – A8) are increasing with the distance from the plasma source as is expected with the $\mathrm{FOM_{rel}}$ factors. Close to the plasma source, for example at locations A1, B1, and in a direct steaming path of one of the ports, e.g. B3, the analog MCNP and AA simulations are similarly efficient.

The FOM_{rel} factors for the SE chimney positions (B1 – B7) are more surprising, especially for locations B5, B6 and B7. The apparent lesser efficiency of ADVANTG to speed-up simulations at those tally locations might be explained by one of the neutron pathways being neglected in the deterministic calculations or perhaps it is caused by the high efficiency of the point-detector tallies for such problems where numerous scatterings of the particle occur near the tallies of interest. Further analyses are needed to better understand this behavior.

Nonetheless, ADVANTG speeds up all tallies of interest, especially when comparing analog simulations with the volume averaged track length estimator tally (F4) where the FOM_{rel} factor is practically infinite because no tally scores occurred in the analog simulations.

Table 1: Results of the Figure-of-merit (FOM) statistical test for the ADVANTG accelerated (FOM_{AA}) and Analog (FOM_{Analog}) MCNP simulations. The ratio of the two is given in the last two columns as the relative Figure-of-merit (FOM_{rel}).

TLD position	**FOM of analog simulation (FOM_{Analog})**		**FOM of ADVANTG accelerated simulation (FOM_{AA})**		**Relative FOM (FOM_{rel})**	
	F4 tally	F5 tally	F4 tally	F5 tally	F4 tally	F5 tally
A1	0	1.80E+00	3.90E-04	1.80E+00	N.A.	1
A8	0	6.70E-02	1.00E-03	1.60E+00	N.A.	24
A2	0	2.60E-02	1.30E-03	8.80E-01	N.A.	34
A3	0	9.10E-03	1.90E-03	1.00E+00	N.A.	110
A4	0	1.20E-02	6.50E-04	1.00E+00	N.A.	83
A5	0	1.50E-03	3.40E-02	7.30E-01	N.A.	487
A6	0	2.10E-04	4.60E-02	1.50E-01	N.A.	714
A7	0	1.60E-04	5.40E-02	1.80E-01	N.A.	1125
B1	0	3.70E+00	1.70E-02	6.20E+00	N.A.	2
B2	0	1.60E-02	8.40E-04	1.10E+00	N.A.	69
B3	0	5.90E-01	1.10E-04	4.20E-01	N.A.	1
B4	0	2.70E-01	7.00E-04	1.10E+00	N.A.	4
B5	0	4.30E-04	2.80E-02	2.40E-01	N.A.	558
B6	0	1.90E-02	1.70E-03	8.50E-01	N.A.	45
B7	0	1.40E-03	8.60E-04	2.30E-01	N.A.	164

6. CONCLUSION

ADVANTG was successfully used to accelerate Monte Carlo MCNP numerical simulations of the neutron fluence at experimental positions throughout the JET tokamak hall. The JET3 NEXP streaming benchmark experiment was described in the paper including the locations and the composition of the TLD assemblies. The effect of three most important ADVANTG input parameters on the final results was presented. The results included a comparison of the convergence of the neutron fluence and the FOMs when using different multigroup data libraries, quadrature sets, and geometrical meshes. The results show that no bias is introduced into the results of the calculation when using various different input parameters. The effects on the convergence of the FOMs show that as expected the dedicated FENDL 3.1b based libraries perform the best alongside with detailed quadrature set and geometrical meshes. The difficulty is to find a balance between them while being limited by the amount of available computer memory.

The parameter variation study shows us the speed up of the simulation differs by approximately a factor of 2 when varying the input parameters. When performing long simulation this is a significant increase. Additionally such an analysis gives a more thorough understanding of the neutron transport process and knowledge which is transferable to following analysis of fusion streaming and shielding problems. One has to be aware that if, for example, a uniform geometrical mesh for the deterministic calculation would be used on a complex fusion problem the results of the following Monte Carlo simulation would be non-physical and inefficient.

Experimental positions in the SW labyrinth (A1 – A8) and in the SE chimney (B1 – B7) have been analyzed with the two-step hybrid deterministic/Monte Carlo method. In order to validate ADVANTG on this complex fusion streaming problem, three different institutions (JSI, ORNL and CCFE) produced separate variance reduction parameters with ADVANTG for each of the experimental positions based on a common full 360^{o} MCNP input file of the JET tokamak. ADVANTG accelerated (AA) results were firstly compared to the analog MCNP calculations. The results agreed within the statistical uncertainty at the locations where the analog results had an uncertainty below 5 %.

The numerical simulations of the three institutions were also compared to the experimental results. The C/E values increase farther away from the plasma with large experimental uncertainties for locations farther away from the plasma. The C/E values range from 0.7 close to the plasma source to 13 in the farthest locations. The cause of the discrepancy between the calculation and experiment can now be analyzed much faster thanks to ADVANTG produced variance reduction parameters.

Finally, the FOM values of the AA and analog calculations were compared in order to determine the relative Figure-of-merit (FOM_{rel}). The factor for the volume averaged track length estimator tallies (F4) were infinity because the analog simulations did not produce any tally scores. The FOM_{rel} factor of the point detector tallies (F5) ranges from 1 close to the plasma source to about 1000 in the SW labyrinth.

ADVANTG has proven to be effective and reliable to use for accelerating complex fusion neutron streaming problems. It does not introduce a bias into the calculations and is user-friendly. The most time-consuming part needed to define the input parameters for ADVANTG is the definition of the geometrical mesh. Development of an alternative method to the FW-CADIS method for the determination of variance reduction parameters for Monte Carlo simulations based on deterministic calculations is planned.

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Appendix A: TLD assemblies locations

Table 2: Locations of A1 – A8 and B1 – B8 TLD assemblies in the JET tokamak Torus Hall according to the MCNP model.

TLD	Coordinates [cm]			TLD	Coordinates [cm]		
	X	Y	Z		X	Y	Z
A1	-672	608	-157	B1	-601	-500	0.
A2	-	1680	-	B2	-1310	-	-
A3	-	1130	-376	B3	-1350	-	-
A4	-	1433	-	B4	-1510	-	-
A5	-	1731	-	B5	-1510	-	-
A6	-	1654.	-	B6	-1415	-	-
A7	-	1920	-393	B7	-	-	-
A8	-896	784	-8	B8	-1605	216	-

Appendix B: Energy dependent TLD attenuation factors

Table 3: Energy dependent TLD attenuation factors that are used to account for the lack of the actual high density polyethylene TLD holders in the MCNP model.

Neutron energy [MeV]	TLD attenuation factor	Neutron energy [MeV]	TLD attenuation factor
2.5000E-08	7.5055E-02	6.0000E-02	2.7888E-01
4.0000E-07	1.1605E-01	7.0000E-02	2.8533E-01
1.0000E-05	1.7418E-01	8.0000E-02	3.0982E-01
1.0000E-03	2.1127E-01	9.0000E-02	3.0495E-01
2.0000E-03	2.2224E-01	1.0000E-01	3.2149E-01
3.0000E-03	2.2654E-01	2.0000E-01	3.4342E-01
4.0000E-03	2.2654E-01	3.0000E-01	4.3449E-01
5.0000E-03	2.2748E-01	4.0000E-01	5.0007E-01
6.0000E-03	2.2679E-01	5.0000E-01	5.7381E-01
7.0000E-03	2.2782E-01	6.0000E-01	6.6360E-01
8.0000E-03	2.3035E-01	7.0000E-01	7.2119E-01
9.0000E-03	2.4034E-01	8.0000E-01	7.8630E-01
1.0000E-02	2.4831E-01	9.0000E-01	8.4616E-01
2.0000E-02	2.5033E-01	1.0000E+00	8.9744E-01
3.0000E-02	2.5033E-01	2.0000E+00	1.2035E+00
4.0000E-02	2.6207E-01	3.0000E+00	1.5302E+00
5.0000E-02	2.6608E-01	1.4000E+01	1.7734E+00