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GLOBAL FLUX CALCULATION FOR IFMIF-DONES TEST CELL USING ADVANCED VARIANCE REDUCTION TECHNIQUE

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ABSTRACT

Global neutron flux and dose maps for the test cell (TC) of the IFMIF-DONES (International Fusion Materials Irradiation Facility- DEMO Oriented NEutron Source) have been calculated applying the advanced variance reduction tool ADVANTG. The neutronics model of the TC has been updated to the current IFMIF-DONES design of the target assembly, the high flux test module, the lithium quench tank as well as the TC surrounding rooms. A weight-window mesh has been produced using ADVANTG with the well-configured setups for IFMIF-DONES conditions. This WW mesh has been adjusted to achieve a reasonable good statistics for the global flux mesh tally. It is concluded that the thickness of the beam upstream and lateral wall can be reduced with 0.5 m without strongly affecting the shielding performance. The neutron streaming through the gaps of the shielding plugs to the access cell above the TC requires local shielding to allow frequent access during operation.

Key Words: IFMIF-DONES, ADVANTG, MCNP, shielding, variance reduction

1 INTRODUCTION

IFMIF-DONES (International Fusion Materials Irradiation Facility- DEMO Oriented NEutron Source, brief as DONES) [1, 2] is a downscaled IFMIF-based neutron irradiation facility which aims at providing the irradiation data required for the construction of a DEMO fusion power plant. DONES consists of only one of the IFMIF accelerators (40 MeV and 125 mA), and utilizes only the High Flux Test Module (HFTM) for the irradiation of material specimens. This facility produces intense neutrons with a neutron flux up to 10^{14} n/cm^2/s and energy up to 55 MeV. The shielding of the facility is very important for the test cell (TC) design. The DONES TC design is based on the design developed during the IFMIF Engineering Validation and Engineering Design Activities (IFMIF/EVEDA) phase. The heavy concrete bio-shield of the TC housing the irradiation test modules is up to 4 m thick. The biological dose rate distribution outside the bio-shield needs to be known with sufficient accuracy for assessing the accessibility during operation and maintenance periods of the facility.

The computational assessment of the radiation penetrating the thick bio-shield is, however, a great challenge for transport simulations. This applies in particular for the Monte Carlo (MC) particle transport technique as available with codes like MCNP5 [3]. Sophisticated variance reduction techniques are required to obtain a sufficient good statistics for the responses scored behind the massive bulk shield. Applying advanced weight-window (WW) mesh generation tools such as ADVANTG [4] for variance reduction of the DONES shielding analyses has been studied in the previous work [5]. Although good acceleration effects have been obtained, accurate global flux and dose maps are still not achieved due to high statistical error in some region away from the TC center. Meanwhile the long neutron history problem, which is caused by over-splitting neutrons t due to impropriate low WW values, is a problem that still appears in new calculation cases.

The global flux and dose maps are essential for the safety classifications of the TC surrounding rooms, as well as reducing the construction cost by optimizing the shielding design. In this work, the application of ADVANTG for variance reductions of DONES MC simulations has been further investigated, aiming at obtaining accurate global flux and dose maps.

2 COMPUTATION MODEL AND METHODOLOGY

The neutronics model, i.e. MCNP geometry, of the TC tagged with a version “mdl8.2.0” has been adopted for this calculation. It is based on the IFMIF/EVEDA design [6] of the TC geometry, with many design changes applied for the DONES TC since then. The first change is the replacement of the target assembly (TA) from the integral back-plate concept proposed by Japan to a bayonet back-plate concept developed at ENEA [7]. The second change is the update with the latest DONES HFTM design [8], which has nearly doubled the thickness in beam direction. Because the Medium Flux Test Module and Low Flux Test Module, designed for IFMIF in the EVEDA phase, are not included in the IFMIF-DONES anymore, the increase of the HFTM thickness provides more irradiation volume as well as a good neutron reflection to the irradiation samples. The comparison of IFMIF/EVEDA and DONES for the TA and HFTM is shown in Figure 1. The third change is the lithium quench tank, located initially below the TC floor. It is placed now inside the TC to avoid strong neutron streaming to the Li Loop Cell (LLC) below the TC floor. Vertical and horizontal cut-views of two versions of TC are shown in Figure 2 and 3.

In addition, the DONES neutronics TC model includes the TC surrounding cells, which include LLC below the TC, Access Cell above the TC, and other the surrounding cells as shown in Figure 4. However, the surrounding cells are voided for the calculations in this work, assuming that the neutron reflecting form these cells back to the TC wall is reasonably low. The target interface room (TIR) neighboring the TC beam upstream wall is not modeled, because the back scattered neutron from the target to the TIR is considered in the ongoing accelerator shielding analysis.

As it is shown in Figure 2 and 3, the DONES TC has shielding walls with 2~4 m in thickness, and also has two empty beam ducts (one of them might be filled with shielding plugs in the future), which give rise to strong neutrons leakage out to TIR. As pointed out in [5] on analog calculation, statistics is very difficult to be improved for more than 1.5 m of the wall. It is a challenging model which required good variance reduction technique for simulating the full map of flux and dose distribution.

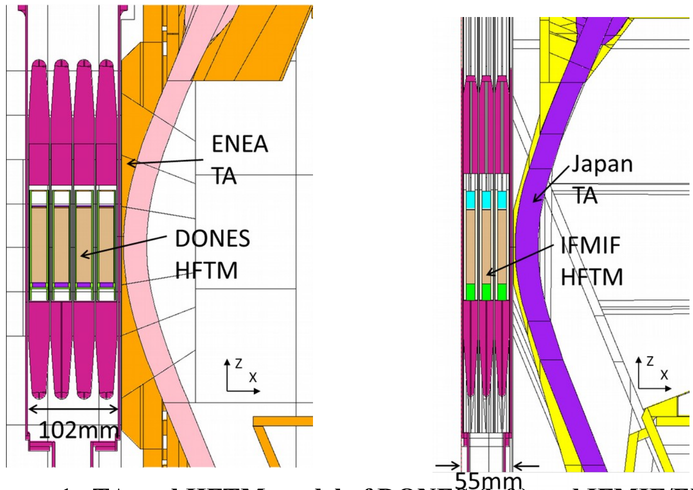

Figure 1. TA and HFTM model of DONES (left) and IFMIF/EVEDA, visual comparison in vertical cut-view.

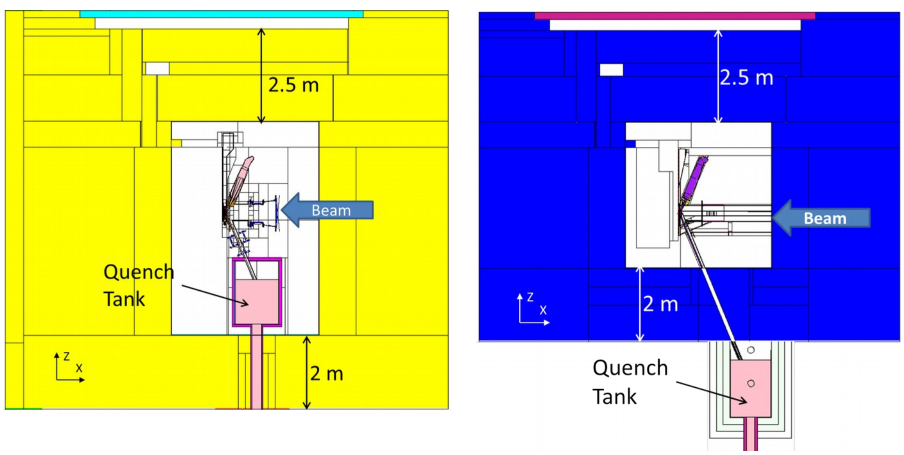

Figure 2. DONES TC (left) and IFMIF TC (right) in vertical cut view at target center.

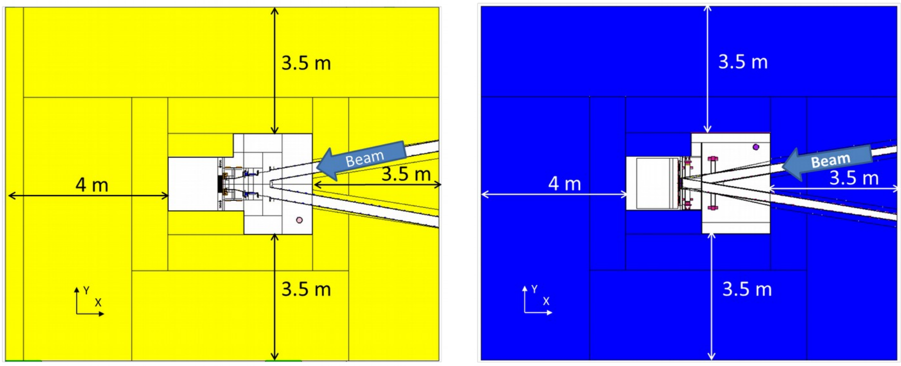

Figure 3. DONES TC (left) and IFMIF TC (right) in horizontal cut view beam level.

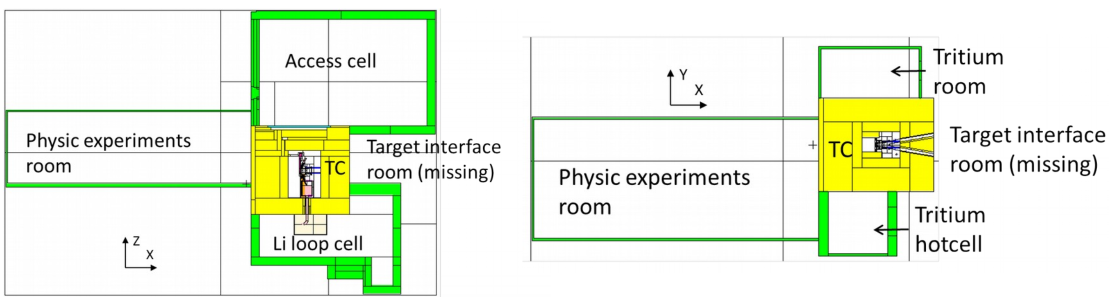

Figure 4. DONES TC surrounding cells in vertical cut (left) and horizontal cut (right)

The ADVANTG code is an advanced WW mesh generation tools developed at Oak Ridge National Laboratory, US. ADVANTG code solves the Boltzmann transport equation using deterministic methods, and then generates WW mesh using the adjoint flux as the important function. It directly processes the geometries and materials in a MCNP input file, and maps them into the specified rectilinear meshes covering the entire model. Also, it builds the source distribution directly from the MCNP SDEF source card, and produces source bias functions which work synergistically with its WW mesh.

For the DONES TC calculation, the deuteron-lithium neutron source, i.e. McDeLicious [9], is implemented as a source subroutine for MCNP. It utilizes external cross-section data files for the simulation of the interaction of deuterons and lithium nuclei. However, the ADVANTG code cannot use the source subroutine for the source modeling. To generate the WW, a SDEF card has been created to approximate the source produced by the McDeLicious subroutine. This SDEF source region is defined in a box covering the deuteron-lithium interaction volume. The uncollided neutron spectrum and angular distribution are sampled in a group-wise structure. However, the source produced by McDeLicious code has also a distribution of source particle weights, while the MCNP SDEF card can only input one weight value. In this calculation, the average source neutron weight are calculated and used as approximation. Such approximations are not critical, since the flux calculated in the ADVANTG code is a priori flux, i.e. approximate flux for generating the WW.

A rectilinear mesh has been set up, which has 91 × 114 × 83 intervals in X, Y and Z directions. In this mesh, fine resolutions up to 10 × 3 × 2 cm are assigned for the region of the target assembly and the HFTM, as well as the beam duct and the Li channel in order to calculate more accurately the streaming neutron flux. In other regions a normal resolution of 20 × 20 × 20 cm is used to limit the size of WW mesh file. The general purpose shielding library 27n19g provided with the ADVANTG package was adopted, which has 27 neutron groups. The FW-CADIS method with the global weighting treatment, which is suitable for generating WW for global mesh tallies, was adopted with 5 order of the Legendre scattering-angle expansion. Since the source bias function produced by ADVANTG cannot be used for the McDeLicious source subroutine, the source bias option of ADVANTG has been turned off. The parallel computation mode has been turned on to enable the use of in total 128 processors (8 × 16 blocks in X and Y mesh direction) for the deterministic transport calculation phase. An important setup for the angular quadrature is, the polar angle is pointing to the X direction in which neutrons largely travel. The maximum of polar angles 16 and the azimuthal angle 37 are used, so that the anisotropic flux distribution can be more accurately simulated in ADVANTG.

3 RESULTS AND ANALYSES

The acceleration effect of the ADVANTG WW mesh has been already detail discussed in [5], thus is not presented again in this paper. The aim of this work is to calculate accurate global neutron flux and dose maps for the safety evaluation. The computational expense is not of greatly concerned, as long as results are good. To meet the needs of heavy computation in this work, the supercomputer MARCONI hosted by CINECA in Italy has been used.

Calculations were made with 10^8 neutron histories for test the effect of the ADVANTG WW mesh, and the statistical errors of the flux mesh tally are shown in Figure 5. By using the fine angle quadrature, the long-history problem is remarkably mitigated. It means there is no obvious frozen in MCNP calculation process. However, as shown in Figure 5, there is a large region inside the rectangles which show no results, especially the region in the beam upstream (+X) direction. While downstream (-X) direction and the beam duct surrounding has very good statistics. It might due to the deficiency of ADVANTG in simulating highly anisotropic problem, or due to the missing of source bias function which is supposed to be used together with the WW mesh. It can be foreseen that, increasing the number of neutron histories will provide limited help to those region, and the WW mesh has to be adjusted in order to mitigate this issue.

The WW mesh has been adjusted using a python script as discussed in [5]. As shown in Figure 5, the WW value of the regions inside the rectangles has been increased along the directions indicated by the arrows. Consider the sweep distance along a rectangle as a unit distance, a relative distance δ $(0 \le \delta \le 1)$ is calculated to the starting plane. Using a global factor λ, the new WW value W_i' of cell i value is adjusted exponentially by multiplying a factor to its original value W_i:

$$W_i' = W_i \exp(\lambda\delta)$$

Here the factor λ used for the rectangles are the number given in the Figure 5. The factors are chosen empirically by trying several times to get acceptable results.

After the adjustment, the WW mesh has been used to calculate the global flux and dose map with 10^{10} neutron histories. The calculation was completed in 14 hours using 480 processors. The flux map is shown in Figure 6. The results with statistics errors <10% are considered accurate and trustable, those with statistics errors of 10-20% are considered somehow useful, and those >20% are consider not meaningful. From Figure 6 it can be seen that large part of results are trustable and meaningful, except some region in the upstream (+X) direction. It is very difficult to adjust this region, because the neutrons from the beam duct would again produce long histories if the weights are over-tuned near the beam duct.

In Figure 6, it shows that the neutron flux varies more than a factor 10^{12} through the TC concrete wall. The asymmetric of neutron flux distribution cause by using one accelerator with incident angle of 9° becomes not evident after 3 m of the downstream (-X) TC wall, but it is still clear that the neutron flux at the downstream-left (-Y) side is larger than the downstream-right (+Y). The vertical cut-view shows neutron streaming effect along the lithium outlet pipe below the QT, due to the weak shielding effect of 35 mm thick of low-density heat insulator layer. Also, two plumps can be found at the two side of the upper shielding, which is caused by the neutron streaming along the 2 cm gaps around the top shielding plug.

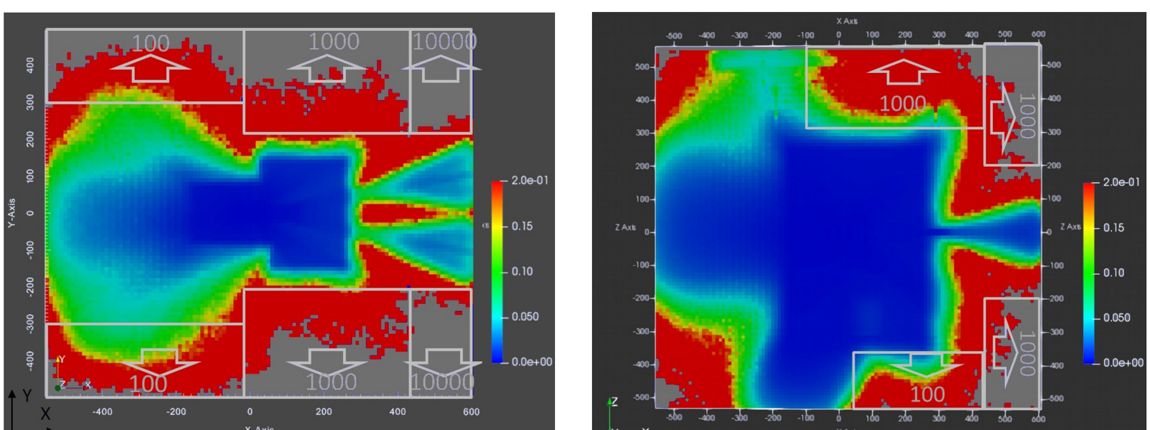

Figure 5. Statistical error of the flux mesh tally in horizontal cut-view (left) and vertical cut-view (right). The number and direction arrow shown in the rectangles indicates the WW adjustments in the later step. In the grey zones no results re obtained.

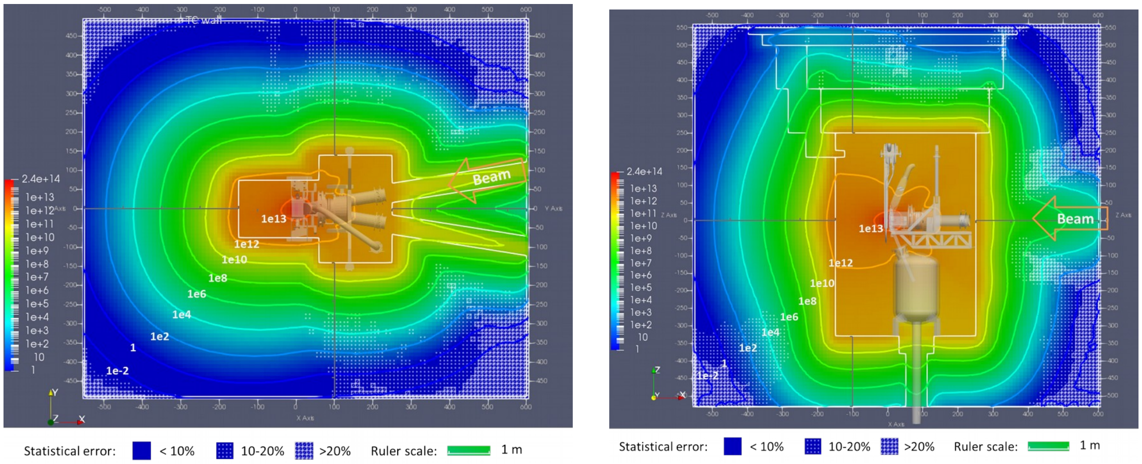

Figure 6. The global neutron flux map of the DONES TC in horizontal cut-view (left) and vertical cut view (right). The statistical error in the region are > 20% with white grids, 10~20% with white dots, and <10% with without marks. The ruler (black line segments) presented in each direction has 1 m for one interval.

Figure 7 shows the neutron dose rate during beam-on. Several colored contour lines are given for the area classification (as shown in Figure 8) defined in the council directive 2013/59/EURATOM [10], which is used as current DONES safety guideline. The TC surrounding cells, except TIR and LLC, are planned to be free accessible during operation. Recently a proposal has been made to reduce the wall thickness in order to reduce the weight as well as the cost of TC shielding. As shown in Figure 7, the downstream (-X), top(+Z) and bottom (-Z) shielding wall are difficult to be reduced since the dose rate to the neighboring cells would increase to the yellow or orange level. The lateral (±Y) wall can be reduced with 0.5 m as the dose value is still within the green level. For the upstream direction the wall can be also reduced with 0.5 m, because large area outside the wall will still in green zone, and the rest will be well shielded by the local shielding inside TIR. An issue is found that the dose rate in the AC partially exceeds to yellow level, due to the neutron streaming along the gaps. Additional local shielding is foreseen at these positions. Note that the photon dose rate is much lower than the neutron dose rate during operation, thus it is not considered in the results.

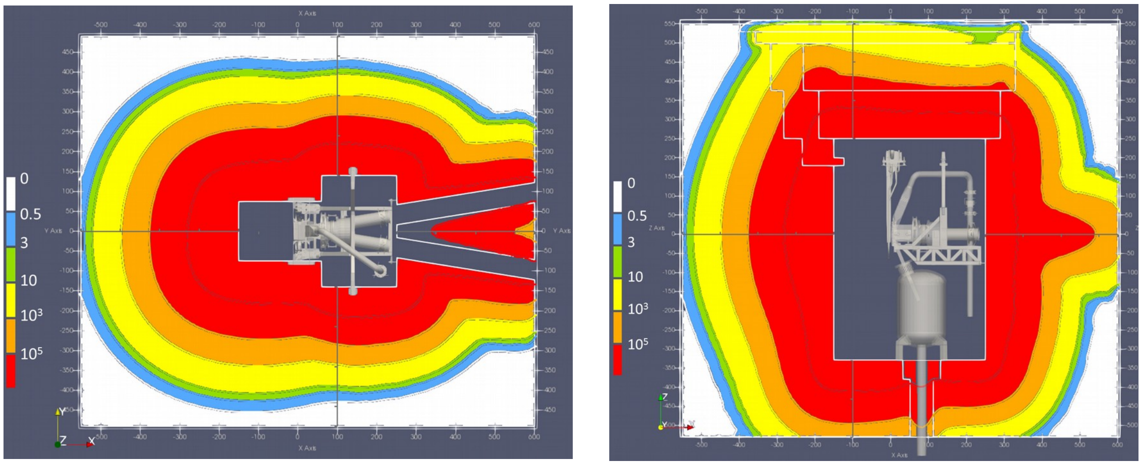

Figure 7. The neutron dose rate (µSv/h) of the DONES TC in horizontal cut-view (left) and vertical cut-view (right). The ruler (black line segments) presented in each direction has 1 m for one interval.

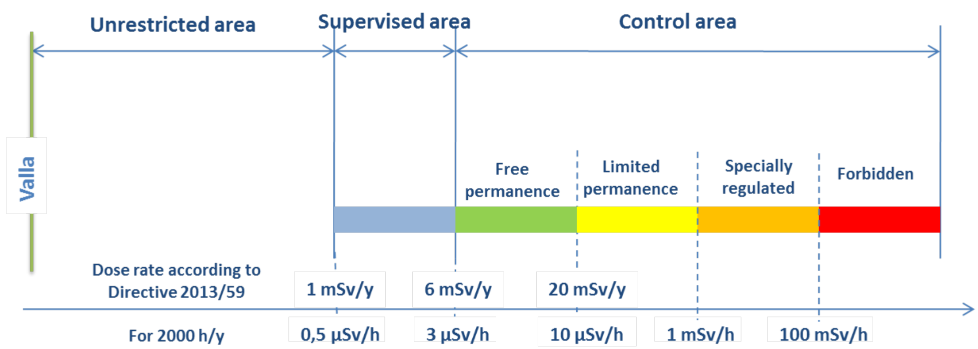

Figure 8. The radiological area classification defined in the council directive 2013/59/EURATOM

4 CONCLUSIONS

Global flux and dose maps of the DONES TC have been calculated applying the advanced variance reduction technique provided with the ADVANTG code. An neutronics model of the TC has been created, which includes new models of the TA, the HFTM, the quench tank and the surrounding rooms. The setup of the ADVANTG calculation has been well-configured for the DONES TC geometry and conditions, although the generated WW mesh still does not enable to get sufficient statistics in some regions. The adjustment of the WW mesh has been discussed in detail, and at the end reasonably accurate global flux and dose maps have been obtained. As result of these calculations, a reduction of the concrete wall by 0.5m in the upstream and the lateral TC walls seem feasible. Such a measure could considerably reduce the construction cost of the TC. The local dose values at the AC partially exceed the allowable limits, thus local shielding need to be applied.

5 ACKNOWLEDGMENTS

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