Decay Dose Shielding Analysis with Hybrid Unstructured Mesh/Constructive Solid Geometry Monte Carlo Calculation and ADVANTG Acceleration

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INTRODUCTION

The Oak Ridge National Laboratory (ORNL) Second Target Station (STS) neutron production facility is an accelerator driven pulsed neutron source that is currently being actively developed at ORNL. The neutrons are produced by proton-induced spallation reactions. A proton beam of 700 kW power is delivered to a spallation target in short, less than 1 μs long pulses, with 15 Hz repetition rate. The spallation target of ORNL STS is a rotating water-cooled tungsten target with tantalum cladding housed in a stainless-steel shroud. It is divided into 21 segments. These segments become highly activated due to spallation reactions or nuclei transmutation by primary protons and emitted neutrons. The radioactive nuclides continue to decay after ceasing operation. The decay dose rates generated from the target segments, once they are removed from their operational location within the core vessel, must be accurately quantified to determine the shielding configurations of remote handling tools and transport casks and to aid in planning maintenance events.

To determine the shielding configurations needed for an activated target segment after ceasing operation, both the hybrid unstructured mesh (UM)/constructive solid geometry (CSG) approach that was previously utilized for STS analyses [1] and the ADVANTG code [2] were used. Even though the ADVANTG code does not include UM capability, the utilization of its advanced variance reduction technique was crucial to accelerate the extremely difficult final photon transport calculation in this analysis. This paper also describes the procedures taken to mitigate the convergence issues that occur when ADVANTG uses a source definition that does not match the source of the final Monte Carlo (MC) calculation. These convergence issues often occur because of inconsistencies between source and transport biasing parameters.

METHODOLOGY

The decay dose rates in spallation environments must be accurately calculated to determine the shielding configurations of remote handling tools and transport casks and to plan maintenance events. A scoping study with a simple geometry was performed to determine these configurations for the irradiated target segment of the ORNL STS. The segment was irradiated using a specific irradiation scenario. It was then placed in thick cylindrical shields and the dose rates were calculated across these geometries.

Calculation Steps

Three calculational steps are required to calculate the decay dose in spallation environments. The goal of the first and second steps is to calculate the decay photon source from the activated components. The goal of the final step is to determine the decay photon dose in a system with various shielding configurations.

Step 1 calculates both the spatial distribution of the neutron fluxes below 20 MeV using MCNP6.2 [3] flux tallies and the radionuclide inventory from protons, neutrons above 20 MeV, and all other particles using the RNUCS card. The RNUCS card is a special tally extension included as an MCNP6.2 patch in the AARE package [4]. It allows tallying the cell-based isotope production and destruction rates using high-energy MCNP6.2 physics models.

Step 2 utilizes the AARE_ACTIVATION script to supply both the neutron fluxes below 20 MeV and the spallation products to the CINDER2008 code. The AARE package includes the CINDER2008 activation code and libraries. The CINDER2008 calculates the decay photon source after a specific operation and decay scenario. In this analysis the operating conditions considered 10 years of operation and 2 weeks cooling down time after ceasing operation. During operation, the beam is considered on for 5,000 hr/year at which the target segment is being irradiated and is considered off for 3,760 hr/year.

Step 3 is a photon transport calculation at which the decay photons calculated in Step 2 are transported through the system to optimize the shielding configurations of remote handling tools and transport casks.

Hybrid UM/CSG approach

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Detailed and complex geometric features inside the target assembly require extremely high-fidelity geometric modeling to assess the effect of key features such as the intricate neutron sources, complex water-cooling channels, and thin layers of cladding.

Step 1 utilized the MCNP6.2 UM capability. This approach is described in Ref. 1. The MCNP6.2 UM model was generated using the Attila4MC [5] code from an original CAD model developed for the target segment. Before converting the CAD model into an UM model, the CAD model was segmented into many cells because this approach requires the segmentation of large components in the original CAD geometry into smaller cells. This segmentation is needed for obtaining a reliable spatial distribution of cell averaged radionuclide inventories using MCNP cell tallies. The segmentation should in principle identify the most important parts of the model that have a high chance to contribute to the activation dose in areas of interest, considering effects such as gamma self-shielding. Figure 1 shows the targe segment after its segmentation into smaller cells for utilizing the cell-based approach in Step 1.

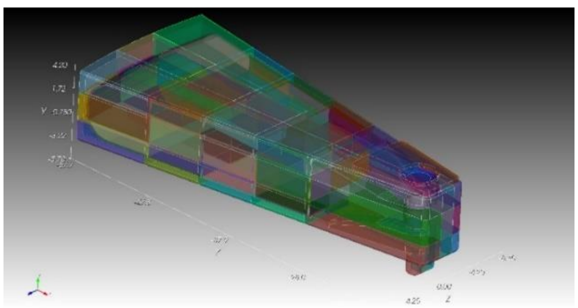

Fig. 1. Target segment after segmentation for utilizing cell-based approach

Step 2 utilizes CINDER2008 to calculate the decay photon source. The total decay photon source was calculated after two weeks cooling time and 10 years of operation. From the CINDER2008 output files, the decay photon source was converted into MCNP6.2 SDEF cards using Version 2.8 of the Gamma Source Perl script released with the AARE package, AARE_GAMMA_SOURCE. The POS parameter of the SDEF cards was set using the *volumer* MCNP6.2 keyword to sample the source uniformly all over volume source regions defined in the segmented cells of the target segment.

The model used in Step 3, the final photon transport calculation, is shown in Fig. 2. It included the target segment enclosed inside a cylindrical shield. The thickness of the shield was 1 m in the radial direction and its base was 20 cm thick. The inner radius of the cylindrical shield was 67.7 cm and the height above its base was 5 m. There was no shield on the top. This geometry was chosen for a scoping study to determine the thicknesses of shielding needed for remote handling equipment and shipping casks. Both Stainless Steel 316 (SS-316) and Lead were used as shield. The standard mcplib84 was used for the MCNP6.2 calculations in this step.

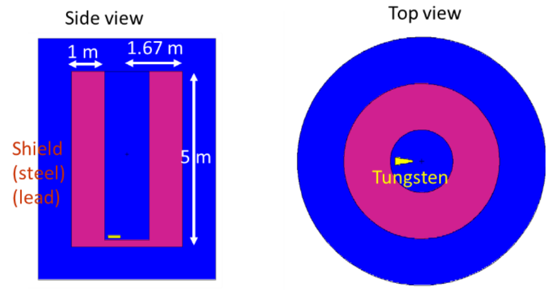

Fig. 2. Geometry of model used for final decay photon transport calculation

ADVANTG calculation

Because very large amounts of shielding were used in this scoping study to determine the thicknesses of shielding needed for remote handling equipment and shipping casks, the MC calculations were only tractable using advanced variance reduction techniques. The steel shield provided more than 25 orders of magnitude attenuation and the lead shield provided more than 30 orders of magnitude attenuation. The ADVANTG code was utilized for this analysis. It generates space- and energy-dependent mesh-based weight-window bounds and biased source distributions using three-dimensional (3-D) discrete ordinates (SN) solutions of the adjoint transport equation that are calculated by the Denovo package. ADVANTG outputs weight-window lower bounds as an MCNP-compatible weight-window input (WWINP) file. Weight-window control parameters and biased source distributions are output as WWP and SB cards, respectively, in an extended version of the user's original MCNP input file [2].

Unfortunately, ADVANTG does not support some key features that are needed in spallation neutron source applications. As described in Ref. [6], some of these limitations can be overcome, some more easily than others. In fact, ADVANTG only supports MCNP5 and does not support different MCNP6 versions. Simplified MCNP5 models were created for ADVANTG calculations in this analysis. The MCNP5 model included the cylindrical shield. The target segment, which was described using the MCNP6.2 UM capability, was approximated by a Tungsten wedge to incorporate its photon attenuation effects in the ADVANTG calculation.

ADVANTG neither supports the MCNP6.2 UM capability nor the MCNP6.2 volume sources that use the *volumer* SDEF card. Defining the MCNP source for ADVANTG is not only used to create a source for the deterministic Denovo calculation, but it is also needed to create the biasing parameters using the Consistent Adjoint Driven Importance Sampling (CADIS) methodology [7].

Without "consistent" source biasing parameters, the source histories of the final MC calculation might suffer from severe over splitting and/or rouletting that could dramatically slow down the MC conversion and make impractical [7]. These source convergence problems become more significant when the source spans over a larger region of energy and space. Because the decay photon source in this analysis has a very detailed energy distribution, the source biasing becomes crucially important for the convergence of the final MC calculation with ADVANTG weight windows. However, calculating accurate source biasing parameters was not possible for this problem because the source described using an UM and the *volumer* SDEF card is not supported by ADVANTG. In this analysis, we approximated the source distribution in both the energy and space domain. The analysis compares the efficiency of the final MCNP6.2 calculation without the use of ADVANTG, with only weight windows created using an ADVANTG calculation that used the MCNP6.2 default photon source, and with both the source biasing and weight window parameters created using an ADVANTG calculation that used the approximate source we created. The MCNP6.2 default photon source is a simple 14 MeV monoenergetic point source. To approximate the source for ADVANTG calculation, all the source probabilities of the small segmentations shown in Fig 1 were averaged. ADVANTG calculation used a volumetric source that covers a wedge similar in shape to the target segment and with average source probabilities. The approximation was only used for the ADVANTG calculation, and the final MCNP6.2 calculation used the actual UM source with the original probabilities and the biasing parameters.

RESULTS

A mesh tally was used to calculate the dose rate map across the entire geometry of heavily shielded activated target segment for both the steel and lead shields. Figure 3 shows a vertical slice for the results of this mesh tally. Only the mesh tally elements that had relative uncertainty < 90% are shown in this Fig. 3. ADVANTG with the approximate source was used for both the calculations shown in Fig. 3. For this paper that focuses on the methodology rather than the shield design, all the dose rate results were normalized.

The 1 m thick steel shield provided 25 orders of magnitude attenuation and the 1 m thick lead shield provided 30. This shows how difficult the shielding problem is, which made the use of advanced variance techniques crucial in this analysis.

Table 1 shows the parameters used for the three MCNP6.2 calculations, namely: MCNP6.2 calculation without ADVANTG, MCNP6.2 with only weight windows from an ADVANTG calculation with a monoenergetic point source, and an MCNP6.2 calculation with both the weight windows and the source biasing parameters calculated using an approximate source.

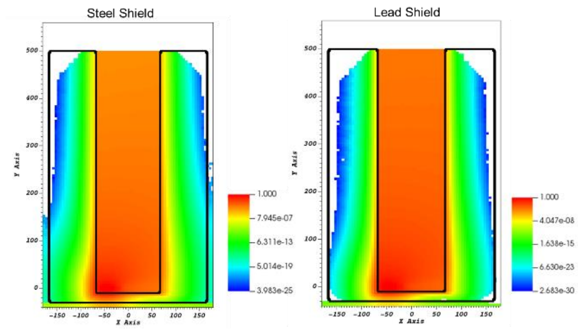

Fig. 3. Normalized dose rates with steel and lead shield

Table 1: Comparison of MCNP6.2 calculations with different ADVANTG options

	Without ADVANTG	With ADVANTG point source	With ADVANTG approximate source
Number of histories	3.60E+10	4.50E+10	6.60E+10
Number of cores	192	192	192
Run time (min)	4.34E+05	5.29E+05	5.35E+05
Percentage of non-zero voxels	17.82%	28.30%	61.88%

Figure 4 shows the dose rate maps calculated using the three methods investigated in this analysis. The results were again normalized. The three calculations used approximately similar times shown in Table 1. Only the mesh tally elements that had relative uncertainty < 90% are shown in this Fig. 4.

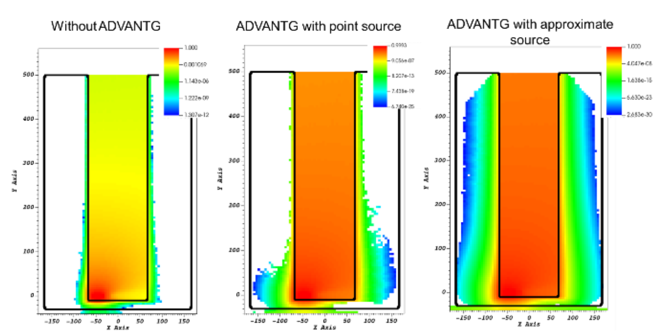

Fig. 4. Normalized dose rates calculated with MCNP6.2 with three different options.

The MCNP6.2 calculation that utilized the weight windows created by an ADVANTG calculation with a point source had ~59% more mesh tally elements (voxels) with non-zero scoring than the calculation that did not use ADVANTG. The calculation that used both the weight window and the source biasing parameters from an ADVANTG calculation with an approximate source had ~3.5 times more non-zero voxels than the calculation without ADVANTG.

Figure 5 shows the relative uncertainties in the dose rate maps calculated using the three methods investigated in this analysis. The relative uncertainties of the zero-scoring elements were assigned 100% for illustration purposes. The MCNP6.2 calculation with both the weight windows and source biasing parameters had low relative uncertainties in the important regions of the thick shield. This enabled the scoping study to determine the shielding thicknesses required to decrease the dose rates to acceptable limits.

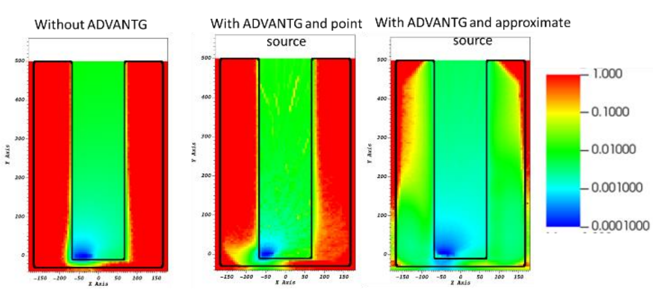

Fig. 5. Relative uncertainties in dose rates for the three different MCNP6.2 calculations.

Figure 6 shows the cumulative distribution function of mesh tally uncertainties for the three MCNP6.2 calculations with different options. The three calculations used approximately similar times shown in Table 1.

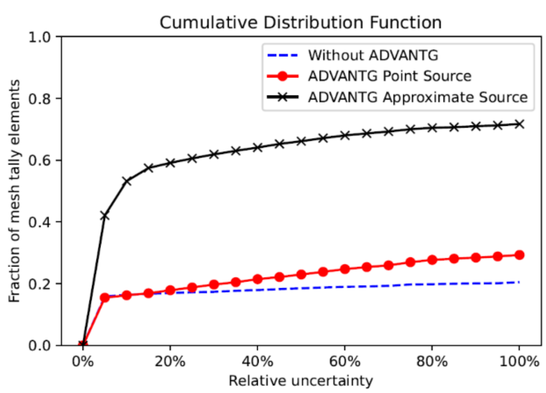

Fig. 6. Cumulative distribution function for mesh tally uncertainties of three MCNP6.2 calculations with different options

At any clear level of relative uncertainty, the MC calculation that utilized both the weight window and the source biasing parameters from an ADVANTG calculation with an approximate source had a higher fraction of voxels with relative uncertainties below this level. This clearly shows that this calculation with an approximate ADVANTG source is much more efficient than the other two calculations.

CONCLUSION

A hybrid UM/ CSG approach was used with ADVANTG acceleration to determine the shielding configurations needed for an activated target segment after ceasing operation. The hybrid UM/ CSG approach allowed combining the UM created for the complex CAD model of the target segment with a simple geometry of the thick shield used for a scoping study.

The decay photon transport calculation through the thick shield exhibited between 25 and 30 orders of magnitude attenuation depending on the shield. Such a difficult shielding calculation required advanced variance reduction. ADVANTG has some missing features, which limits its usability in spallation neutron source applications [6]. It does not support volumetric sources created for MCNP6.2 UM capability. An approximate source was created for this problem. This approximate source was not just needed for running the ADVANTG calculation to generate the weight windows, it was also essential to develop source biasing parameters that were crucial for dramatically accelerating the decay photon transport in this problem. Investigating whether ADVANTG capabilities could be expanded for these types of spallation neutron applications and/or it should be replaced by Attila4MC [5] is underway.

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