

Task 42

Monaco/MAVRIC Evaluation for Facility Shielding and Dose Rate Analysis

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BACKGROUND

The dimensions and the large amount of shielding required for Global Nuclear Energy Partnership (GNEP) facilities, advanced radiation shielding, and dose computation techniques are beyond today's capabilities and will certainly be required. With the Generation IV Nuclear Energy System Initiative, it will become increasingly important to be able to accurately model advanced Boiling Water Reactor and Pressurized Water Reactor facilities, and to calculate dose rates at all locations within a containment (e.g., resulting from radiations from the reactor as well as the from the primary coolant loop) and adjoining structures (e.g., from the spent fuel pool).

In complex geometry problems, Monte Carlo methods are often used to compute fluxes or dose rates over large areas using mesh tallies. For problems that demand that the uncertainty in each mesh cell be less than some set maximum, computation time is controlled by the cell with the largest uncertainty. This issue becomes quite troublesome in deep-penetration problems, and advanced variance reduction techniques are required to obtain reasonable uncertainties over large areas. To overcome this issue, Oak Ridge National Laboratory (ORNL) has developed a new sequence, MAVRIC, which will be available with the release of SCALE 6. In this sequence, a methodology called Consistent Adjoint-Driven Importance Sampling (CADIS) has been incorporated for effective variance reduction. This was developed to quickly and automatically determine the biased source distribution and weight windows over a rectangular mesh and a given energy group structure. The method first determines the approximate adjoint particle flux, usually using a discrete ordinates code. The source for this calculation is the detector energy-group response for the process of interest (e.g., dose rate) at the location(s) of interest. The resulting adjoint flux at each location and energy is equated to the importance of particles and is combined with the source distribution to generate the biased source and weight window values that control particle populations at all locations. Very recently, a variation of the CADIS methodology, referred to as the Forward-Weighted CADIS (FW-CADIS) method has been developed, implemented in MAVRIC, and demonstrated for optimization of dose maps.

RESEARCH OBJECTIVES AND METHODS

The MAVRIC sequence is being evaluated along with the Monte Carlo engine MONACO to investigate its effectiveness and use-

ACADEMIC YEAR HIGHLIGHTS

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fulness in facility shielding and dose rate analyses. A previously MCNP-evaluated cask array from the Yucca Mountain Project's proposed aging pad and/or buffer area design will be utilized for evaluation and benchmarking purposes. In addition, dose mapping will be performed inside the surface facilities utilizing a transportation cask to evaluate the effectiveness of the code systems. The ability to calculate doses in deep-penetration problems will also be evaluated.

RESEARCH ACCOMPLISHMENTS

The project was initiated in collaboration with ORNL. The first step was initial acquisition of the MONACO/MAVRIC code system from ORNL and identification of a cask array configuration for evaluating the code. 3-D importance/tally mesh was optimized for neutron-photon transport simulation of a single used fuel aging cask. This configuration was modeled in MAVRIC and the model was transmitted to ORNL for review. In addition, inputs were established for a variety of 4x4 cask arrays to match cask storage configurations anticipated at the Yucca Mountain Project. Analyses for the 4x4 aging cask arrays were optimized with respect to dose analysis in between and far from the casks. In addition, the practicality of analyzing a 12 x (4x4) aging cask array is being evaluated with respect to memory limitations.

Other progress:

- Previously encountered memory issues with the SCALE/MONACO/MAVRIC code were resolved by ORNL.
- Dose assessment of a single aging cask was completed. The data indicates excellent agreement to previous data obtained analyzing the same geometry with the MCNP code.
- Complex cask geometry incorporating air vents has been evaluated. As anticipated, the dose rate profile around the cask is slightly different when compared to a simple geometry, and higher dose rates are observed close to the air vents just outside the cask.
- Neutron-neutron and neutron-photon source-response analysis is being examined to determine if it significantly alters the results for much quicker photon-photon analysis.

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