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Radiation dose rate distributions of spent fuel dry casks estimated with MAVRIC based on detailed geometry and continuous-energy models

Yuan Gao *, Christopher R. Hughes, Christopher R. Greulich, James E. Tulenko, Andreas Enqvist, James E. Baciak

Nuclear Engineering Program, University of Florida, 100 Rhines Hall, Gainesville, FL 32611, United States

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ABSTRACT

This study presents a detailed comparison of the dose rate distributions of a dry TN-32 fuel cask with two geometry models and two cross-sectional datasets. The accuracies of radiation dose rate estimation and computational efficiencies of each geometry model with two cross-sectional data-sets are compared. The use of automated variance reduction techniques can significantly improve the computational efficiency of such a realistic, deep penetration problem that involves radiation transport from different volumetric sources, thereby eliciting only a small statistical error. Monaco with Automated Variance Reduction using Importance Calculations (MAVRIC) is a computational sequence within the SCALE 6.2 code package based on consistent adjoint driven importance sampling (CADIS), a type of automated variance reduction technique. Homogenous and full fuel assembly models are built herein, and two nuclear cross-section libraries (V7-200N47G and continuous energy) are applied in this work. Based on the detailed comparisons, we found that neutron dose rate estimation is more dependent on geometry modeling than on cross-section data. For neutron-induced gamma rays, the dose rate distribution depends on both the spatial self-shielding effect and the cross-section library. The primary gamma rays respectively contribute to the total dose rate by ~91% and ~99% on the side and top surfaces, and the dose rate accuracy is more dependent on the cross-section library than on geometry modeling. In terms of the computation efficiency and efforts spent on geometry modeling, the homogenous fuel assembly model with the MG library can produce an acceptable dose rate distribution, but the detailed fuel assembly model with the continuous-energy library is required for more precise dose rate estimation.

1. Introduction

Dry storage casks for spent nuclear fuels were first employed in 1986 at the Surry Nuclear plant as a temporary solution for high level nuclear waste storage until the identification of a permanent solution. However, no permanent solution has been finalized or put in operation for many of the world's nuclear power plants since the time dry cask storage was first used. Therefore, dry storage casks have become a semi-permanent solution rather than a temporary one. Additionally, the number of loaded dry storage casks has been increasing dramatically. By year 2013, approximately 2300 dry storage casks were loaded, and it is estimated that 4500 dry storage casks will be loaded by 2025 (U.S.DOE, FCRD–NFST, 2013). Because of the increased service life and rapid growth of loaded dry storage casks, a concern has been raised about the safety and integrity of their internal structures. There are no internal sensors to monitor the structures placed in the casks and the casks are not easy to open owing to their highly radioactive waste contents. Therefore, the DOE has funded research projects that have focused on the nondestructive evaluation (NDE) of spent fuel casks. One such project had a set objective to investigate and monitor the condition of spent fuel containers, and comprised a research network that included four universities, one spent fuel cask manufacturer representative from AREVA, and one utility representative from EPRI. The set objective of the University of Florida NDE team was to use emission source tomography to determine and verify the spent fuel location and condition.

Emission source tomography is based on the radiation signal emitted from the object itself. Research involving radiative signatures and radiation shielding analyses constitute the fundamental components of our objective. The shielding analysis of the cask can verify the radiation safety of the currently used dry storage cask models. Among the various existing cask models, the transnuclear TN-32 was chosen as the investigating target, which is one of the noncanistered spent nuclear fuel (SNF) storage casks used in

* Corresponding author.
E-mail address: gyizx@ufl.edu (Y. Gao).

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the U.S. Dry fuel cask radiation shielding analyses are associated with the challenging problems of deep penetration and radiation transport from volumetric sources. They have been studied by several researchers in the past few years. Among these studies, the deterministic (discrete ordinates) and stochastic (Monte Carlo) methods have been the most utilized methods.

The Monte Carlo method is considered as the most accurate method, but it requires increased computational cost. Automated variance reduction techniques that apply the deterministic adjoint function in the Monte Carlo code can improve the computation efficiency (Wagner and Haghighat, 1998). The consistent adjoint driven importance sampling (CADIS) methodology is one of the major achievements in the field of Monte Carlo variance reduction. Haghighat and Wagner (2003) illustrated their CADIS methodology and presented the concept of "importance." They performed a gamma-ray dose rate calculation at a storage cask side with the A^3 MCNP code, and showed that the A^3 MCNP code increased the figure of merit (FOM) by a factor of more than 2000 compared with the unbiased MCNP code simulation. This result illustrated the advantage of the variance reduction technique. Therefore, Monte Carlo codes with implemented automatic variance reduction techniques constitute the best choice for this work. Recently, the new SCALE6.2.1 (Rearden and Jessee, 2016) code system was released, and the radiation computation sequence of Monaco with automated variance reduction using importance calculations (MAVRIC) was updated (Peplow, 2011). MAVRIC can perform radiation deep penetration problems with automatic variance reduction using adjoint biased source and importance map to improve the Monte Carlo computational speed. Compared with its predecessor, the SAS4 sequence (Broadhead et al., 1991), MAVRIC computation results are closer to the measured dose rates because of the three-dimensional variance reduction technique it employs rather than the one-dimensional biasing scheme in SAS4 (Thiele and Borst, 2009). Chen et al. (2011) compared the surface dose rate calculations of a spent fuel storage cask with MAVRIC, SAS4, and MCNP, and concluded that MAVRIC is the most efficient code with a satisfactory uncertainty. In this study, we used the new MAVRIC sequence of SCALE 6.2.1 with the latest ENDF/B-VII.1 cross-section library, based on the advantages outlined above.

The accuracy of a complete shielding analysis for a spent fuel cask depends on both the Monte Carlo transport code and on the cask geometry modeling and radiation source term. In this work, the fuel cask geometry and radiation source term were modeled in detail to predict a realistic dose rate distribution over the cask surface and its surroundings. The increased accuracy of this simulation work will constitute the foundation for our future emission source imaging work. Additionally, the shielding design and radiation analysis of the TN-32 cask was independently verified and validated through this work.

2. Materials and methods

The main purpose of this work is to make a reliable evaluation of a complicated shielding problem. The accuracy of this evaluation is mainly determined by two factors, how accurate we can reproduce all the important geometrical features and the accuracy of the underlying nuclear data, especially for nuclear cross-sections. Two geometry models and two nuclear cross-section libraries are used to identify a good evaluation model that best reflects reality with affordable efforts and computational resources.

2.1. Geometry modeling

The TN-32 dry storage cask was designed to accommodate 32 intact pressurized water reactor assemblies, with or without burnable poison rods assemblies (BPRAs) and thimble plug assemblies (TPAs) (Transnuclear, 1999). To conduct a "benchmark" shielding analysis, neither BPRA nor TPA were considered in this work, but only the standard fuel assemblies were inserted. The overall length of the TN-32 cask, including its protective cover, was 513.75 cm, and had an outside diameter of 248.3 cm, and the total weight was 115.5 tons when it was loaded on the storage pad (Transnuclear, 2002). As shown in Fig. 1, the top part of the cask is a protective cover, and it is not considered in the simulation model. Below the protective cover are 10.2 cm of polypropylene encased in a 0.64 cm (0.25 in) thick steel shell, and 26.7 cm thick steel in the lid, which compromises the top neutron and photon shielding materials. The diameter and length of the main containment vessel are 174.6 cm and 414.7 cm, respectively, and can hold 32, Westinghouse 17 × 17 standard fuel assemblies. Radial photon shielding is provided by the 3.8 cm thick inner steel shell, the 20.3 cm thick steel body wall, and the 1.3 cm thick outer steel shell. The radial neutron shielding is provided by a borated polyester resin compound with a thickness of 11.4 cm (4.5 in) cast into slender aluminum containers that surround the cask body. The bottom shielding of photons is provided by the 3.8 cm (1.5 in) thick steel base of the main containment vessel and a 22.2 cm (8.75 in) thick steel bottom plate. There is no neutron shielding material at the bottom because the cask will be placed vertically on the ground, and a negligible amount of neutrons emitted from the bottom will lead to the emitted dose. The detailed dimensions and material information used in this work were described in Chapter 5 of the Final Safety Analysis Report (FSAR) for TN-32 (Transnuclear, 2002).

The shielding evaluation work in Chapter 5 of FSAR for TN-32 cask (Transnuclear, 2002) was reviewed. Dose rate distributions around the TN-32 were determined with the use of Westinghouse, 17 × 17 Standard fuel assembly incorporated within a three-dimensional SAS4 (shielding analysis No. 4) model (Broadhead et al., 2000). The SAS4 model was divided into two parts from its

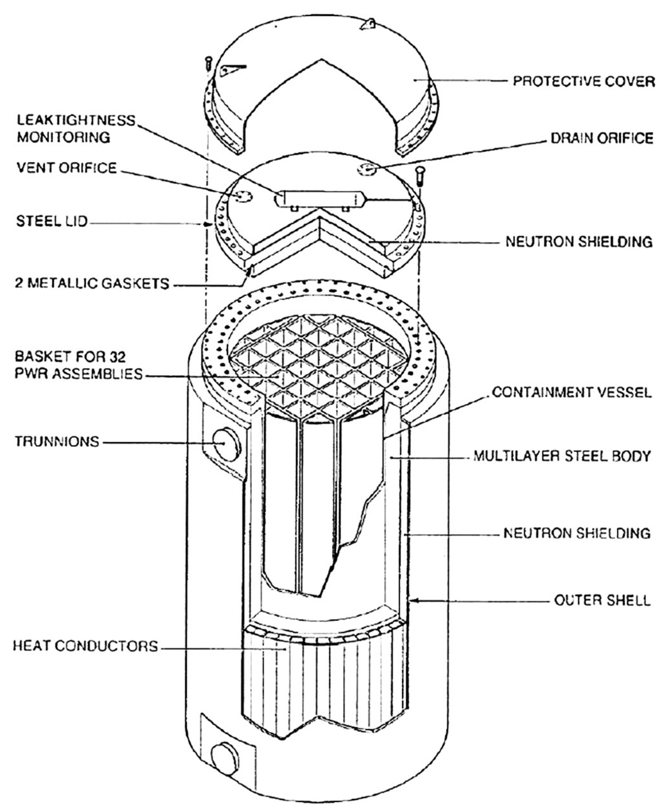

Fig. 1. Exploded/cutaway diagram of the TN-32 cask (Greene et al., 2013).