Surface dose rates of the spent fuel dry storage cask system at the first nuclear power plant in Taiwan

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Abstract: Source terms and surface dose rates of a spent fuel dry storage cask system were evaluated using TRITON and MAVRIC in the SCALE 6.1 code package. The cask system consists of four major components, called the transportable storage canister, transfer cask, vertical concrete cask, and add-on shield. Based on advanced fuel depletion and radiation transport methodologies, source characteristics of the design basis spent fuel and detailed dose rate distributions over the entire surface of the cask system were obtained. The results confirm the appropriateness of the original shielding analysis and demonstrate great advantages of using this approach. In addition, the comprehensive dose rate distributions can provide useful information in preparing associated health physics programs during the transportation and dry storage of spent fuels.

Keyword: spent nuclear fuel; dry storage cask; source term; radiation shielding; Monte Carlo

1 Introduction

The first nuclear power plant (NPP1) in Taiwan has been commercially operated more than 35 years. The capacity of its spent fuel storage pool is nearly exhausted even after two rerackings. The power company has constructed an independent spent fuel storage installation (ISFSI) at the plant site to solve this impending shortage problem. According to its safety analysis report (SAR) [1], neutron and gamma-ray source terms resulting from the design basis spent fuel were estimated using the SAS2H sequence in the SCALE code package version 4.4a [2]. The subsequent shielding calculations under various conditions and scenarios were performed using the Monte Carlo transport code MCNP version 4C [3]. The qualities of these two analyses (source term and shielding) were parts of key issues for the review and licensing of the ISFSI operation. In light of this background, this study repeated the two analyses using independent calculation tools, TRITON and MAVRIC, which are two important control modules in the SCALE code system for reactor physics and shielding applications, respectively. The latest version is SCALE 6.1 [4]. The results were compared with those in the original SAR and advantages of using the new and independent methods were discussed.

2 Materials and methods

2.1 Modeling of the dry storage cask system

The dry storage cask system at the NPP1 consists of four major components, called the transportable storage canister (TSC), transfer cask (TFR), vertical concrete cask (VCC), and add-on shield (AOS). Each TSC can accommodate 56 spent fuel assemblies of the boiling water reactors and the other three components (TFR, VCC, and AOS) have important functions in radiation shielding during the transportation and storage of the TSC. The design basis of spent fuel was a typical GE8×8-1 fuel assembly with a burnup of 36 GWD/MTU and cooling for 10 years. Different initial enrichments were set for conservative source term evaluations. The enrichment of 3.25% was adopted in the estimation of gamma rays from actinides and fission products in a spent fuel, called the fuel gamma source. A relatively low enrichment of 1.9% was used in calculating the fuel neutron source, and also the hardware gamma source mainly caused by neutron activation in fuel structure materials.

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The standard storage cask without AOS is essentially the design of the NAC-UMS system [5], which includes the canister, steel liner, concrete shielding, air inlets/outlets, and other supporting structures. The overall dimensions are approximately 5.7 m in height and 3.4 m in diameter. For detailed dimensions of each component in the dry storage cask system, please refer to Reference [1]. The design of AOS was to enhance the original cask shielding because of a relatively short distance to the nearest site boundary and a stringent dose limit. The modeling of the fuel, TSC, TFR, VCC, and AOS was carefully implemented in this study to ensure consistency with those defined in the SAR. Figure 1 shows the calculation models of four major components of the dry storage cask system, starting from the inner TSC on the left-hand side of the figure to the final configuration with AOS on the right-hand side. To facilitate shielding calculations, the content of the canister (mainly 56 spent fuel assemblies) was homogenized into four source regions: upper end fitting (UEF), plenum, effective fuel region, and lower end fitting (LEF). Fuel neutron and fuel gamma sources were distributed in the effective fuel region with appropriate axial and energy distributions. In addition, there were various hardware gamma sources in all regions, considering the amounts of cobalt impurity in structure materials and neutron flux distribution.

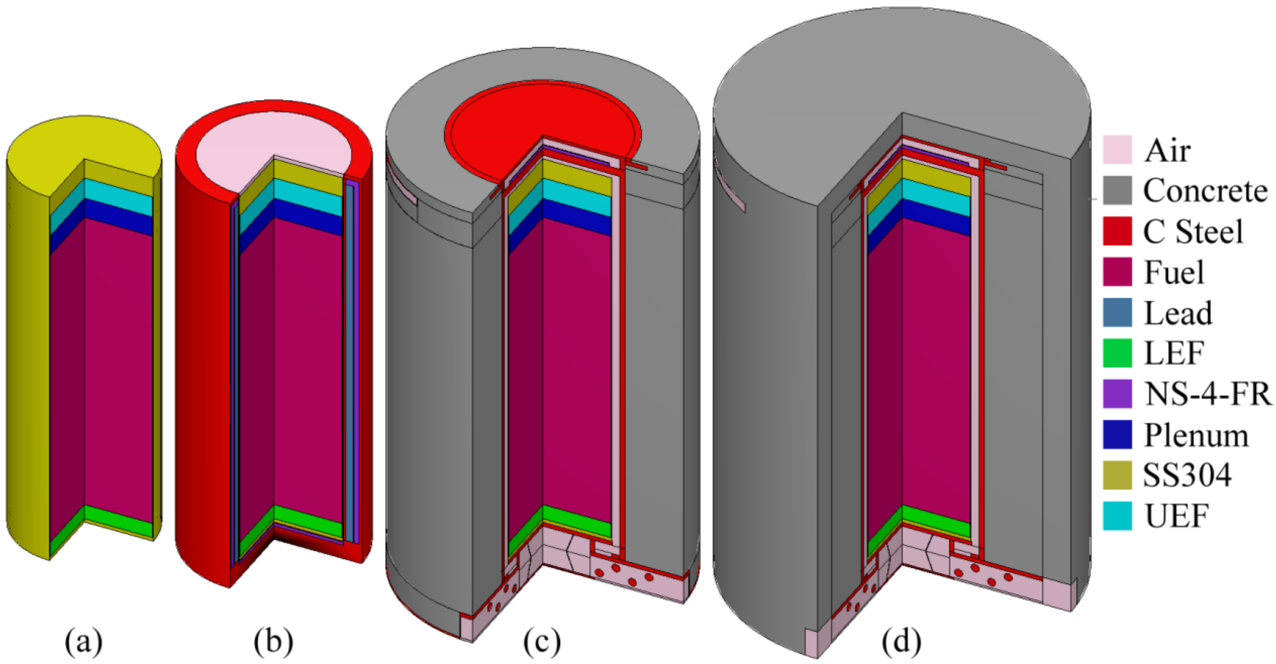

Fig. 1 Geometric models of four major components of the dry storage cask system: (a) TSC, (b) TFR, (c) VCC, and (d) AOS.

2.2 TRITON and MAVRIC

Based on automatic and rigorous treatments of cross-section processing, neutron transport, and fuel depletion, the TRITON module in SCALE can accurately predict the nuclide composition of a spent fuel and associated radiation sources. As a successor to the SAS2H module, TRITON features several key improvements in spent-fuel characterization, such as the advanced problem-dependent cross-section processing and explicit 2D/3D geometrical modeling capabilities. Currently, TRITON only supports multigroup depletion calculations. BONAMI and CENTRM/PMC with the 238-group ENDF/B-VII library were specified in this study for the multigroup cross-section processing to retain maximal accuracy of neutron transport and fuel depletion [4].

Monte Carlo transport calculation for a heavily-shielded spent fuel storage cask essentially involves several computational difficulties, such as complicated geometries, multiple sources, deep penetration, and radiation streaming. The MAVRIC module in SCALE was developed to make challenging Monte Carlo simulations computationally practical by using an effective variance reduction technique called the Consistent

Adjoint Driven Importance Sampling [6]. Detailed models of the four major components of the dry storage cask system (*i.e.*, TSC, TFR, VCC, and AOS) were considered and built in MAVRIC and the 27N-19G ENDF/B-VII shielding library in SCALE was used in the neutron-gamma-coupled transport calculations. In addition to scoring average fluxes on cask surfaces, mesh tallies were used to provide detailed flux distributions over several regions of interest. The same set of fluence-to-dose conversion factors as that used in the original analysis was employed in this study to ensure meaningful comparisons with dose quantities reported in the SAR [1].

3 Results and discussion

3.1 Source terms

New estimates on the intensities and energy distributions of three source terms resulting from the design basis spent fuel were obtained in this study based on the TRITON calculations. Table 1 and Fig. 2 show their comparisons with the original SAS2H analyses. The energy spectra in Fig. 2 indicate that most neutrons are coming from spontaneous fissions of actinides in the spent fuel, and ^{137}Cs and ^{60}Co play important roles in fuel gamma and hardware gamma sources, respectively. The characteristics of neutron and gamma-ray spectra predicted by TRITON and SAS2H appeared to be consistent. In terms of the total intensity, TRITON predicted slightly more fuel neutrons and gamma rays than SAS2H by 8.8% and 4.2%, respectively. However, the predicted hardware gamma rays are approximately 20% lower (Table 1). Despite these minor differences, the comparisons confirmed the appropriateness of the previous source term evaluation. The new estimates are believed more accurate and reliable because of substantial improvements in methodology and data library used in this study. TRITON computed problem-dependent multigroup cross sections through CENTRM/PMC, which in many aspects surpasses the conventional Nordheim Integral Treatment for resonance self-shielding in SAS2H [7]. The new calculations were based on the latest ENDF/B-VII data library in 238 groups, whereas the SAS2H calculations in the SAR were performed with outdated 27BURNUPLIB cross sections. In addition, the design basis BWR fuel assembly was modeled explicitly pin-by-pin in the 2D geometry of TRITON, while SAS2H approximated the problem with a two-level 1D approximation [2].

Table 1 Comparison of three source terms provided in the SAR with those estimated using TRITON in this study for the design basis spent fuel

Source term	SAR (SAS2H)	This study (TRITON)
Fuel neutron (n/s/assembly)	1.268×10^{8}	1.380×10^{8}
Fuel gamma (γ/s/assembly)	1.113×10^{15}	1.160×10^{15}
Hardware gamma (γ/s/kg)	4.804×10^{12}	3.824×10^{12}

3.2 Surface dose rates

For each cask configuration, three separate MAVRIC shielding calculations were carried out corresponding to three types of fixed sources (fuel neutron, fuel gamma, and hardware gamma) derived from the design-basis spent fuel. Figure 3 shows dose rate distributions on all cask surfaces. The average total dose rates on cask surfaces were compared with those reported in the SAR calculated with MCNP. As shown in Table 2, the results of two analyses are rather consistent considering the complexities of the problem and subtle differences in their calculation methodologies between MAVRIC and MCNP.

MCNP is a general-purpose Monte Carlo transport code that features pointwise cross-section data. Continuous-energy Monte Carlo simulations are generally considered the most accurate. However, effective use of variance reduction techniques is crucial for this shielding problem. In order to obtain reliable results within reasonable computing time, MCNP with conventional geometrical importance was the main method used in the SAR. However, the computation efficiency of this scheme heavily depends on the user's experience and judgment. The most important advantage of using MAVRIC in this challenging shielding problem was a profound gain in computation efficiency. The total computational effort for shielding analyses in the SAR was significant [1]. By contrast, all the calculations in this study can be finished within a week on an off-the-shelf personal computer. The key of the enormous speed-up in computation efficiency is that MAVRIC combines the strengths of both the Discrete

Ordinates (S_N) and Monte Carlo transport calculations. The two methods are fundamentally different in many aspects, but they could be used together in a complementary manner. MAVRIC based on the CADIS methodology makes use of the knowledge of an approximate S_N adjoint function to alter the random sampling procedure in the Monte Carlo transport simulation.

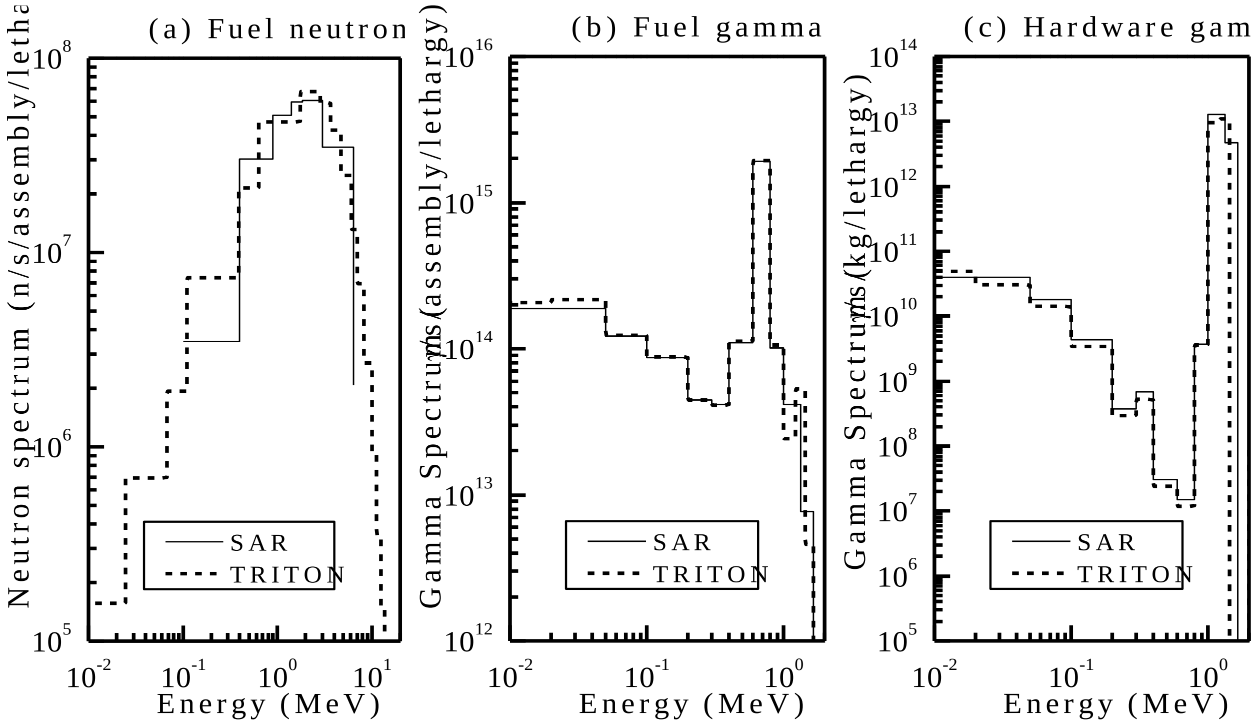

Fig. 2 Comparison of the TRITON-calculated spectra with those presented in the SAR for three source terms of the design basis spent fuel.

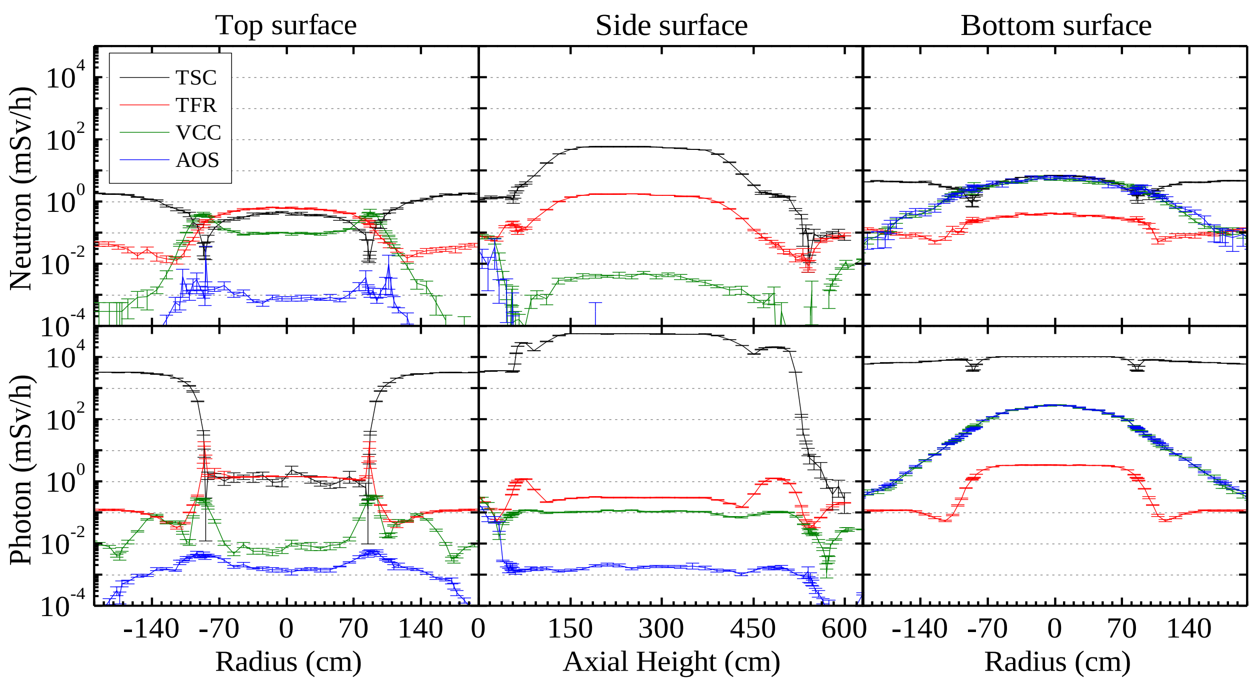

Fig. 3 Neutron and gamma-ray dose rate distributions over the top, side, and bottom surfaces of four major components of the dry storage cask system (TSC, TFR, VCC, and AOS).

Figure 3 gives the resulting neutron and gamma-ray dose rate distributions over the top, side, and bottom surfaces of four major components of the dry storage cask system. Gamma rays dominate the surface dose rates of the TSC. The dose rates are extremely high ($\geq 10^4$ mSv/h) at the side and bottom surfaces, which therefore are not allowed to be exposed in air without shielding. The TFR can effectively attenuate the radiation from the TSC during its transportation from underwater to a dry VCC. Maximal dose rates outside the TFR are at levels of around several mSv/h, where neutrons dominate the dose contribution at the cask side and gamma rays dominate at the bottom surface. For interim dry storage, the TSC is protected and shielded by the VCC. Dose rates at the side surface of the VCC are approximately 0.1 mSv/h that mainly contributed by gamma rays; similar dose rate levels at the top surface are resulted from neutrons. The peak dose rate at the cask top surface corresponds to the location of the air gap between the TSC and the VCC because of radiation streaming.

Table 2 Comparison of average total dose rates (mSv/h) on cask surfaces reported in the SAR with those estimated using MAVRIC in this study for the dry storage cask system

Component	Surface	SAR (MCNP4C)	This study (MAVRIC)
TSC	Top	--	1.487 ±0.01%
	Side	--	42390 ±0.19%
TFR	Top	2.054	1.895 ±1.75%
	Side	3.240	1.662 ±0.15%
VCC	Top	1.92×10^{-1}	1.85×10^{-1} ±1.89%
	Side	1.13×10^{-1}	1.03×10^{-1} ±1.17%
AOS	Top	6.58×10^{-3}	4.12×10^{-3} ±6.87%
	Side	2.45×10^{-3}	1.82×10^{-3} ±3.13%

Radiation shielding of the dry storage cask at the NPP1 will be further enhanced by installing an AOS to comply with a stringent dose limit at site boundaries. Dose rates at the top and side surfaces of the AOS thus become very low, only at levels around 0.01-0.001 mSv/h or below except for the four bottom air inlets of the cask. On the other hand, as shown in Fig. 3, gamma-ray dose rates at the bottom surface of the VCC and AOS could be higher than 100 mSv/h. In practice, this is not an issue because of the storage cask has been designed to prevent from tilting or falling down for all design basis accidents.

4 Conclusion

This study performed an independent verification of the source-term and shielding analyses for the dry storage cask system using TRITON and MAVRIC. TRITON was designed to perform high-fidelity depletion calculations by combining the capabilities of cross-section processing, neutron transport, and fuel burnup. MAVRIC was designed to solve challenging shielding problems by combining the advantages of deterministic and Monte Carlo transport calculations. Based on the state-of-the-art methodologies, source characteristics of the design basis spent fuel and dose rate distributions over cask surfaces of the TSC, TFR, VCC, and AOS were reevaluated. The results confirmed the validity of the original analyses in the SAR. In addition, comprehensive dose rate distributions around four major components of the dry storage cask system can provide useful information in preparing associated health physics programs for future ISFSI operation under normal operation and accident situations.

Acknowledgement

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