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A Comparative Shielding Analysis of the KN-12 Cask with MCNP6 and SCALE6.1

Ye Seul Cho[a], Dae Sik Yook[b], Ara Go[b], Ser Gi Hong[a]*
[a]Department of Nuclear Engineering, Kyunghee Univ., 1732 Deogyeong-daero, Giheung-gu, Yongin, Gyeonggi-do, 446-701, Korea
[b] Korea Institute of Nuclear Safety, 62 Gwahak-ro, Yuseong-gu, Daejeon 305-338, Korea
*Corresponding author: sergihong@khu.ac.kr

1. Introduction

The management of the spent fuels discharge from commercial nuclear power plants is one of the most urgent problems to be resolved in nuclear industry. In particular, the capacities of the current spent fuel pool storages for PWR spent fuels are expected to be saturated in the near future. So, the dry storage of the PWR spent fuels are seriously considered in our country. The cask has an important role in the management of spent fuel for transportation and storage. The KN-12 cask was designed and manufactured to transport 12 PWR spent fuel under dry and wet conditions. This cask is designed to load the PWR spent fuels of burnup less than 50,000 MWD/MTU, initial uranium enrichment below 5.0wt%, and cooling time longer than 7 years. It was licensed in 2002 and owned by Korea Hydro & Nuclear Power [1]. In our country, the evaluation of shielding problem is mostly carried out using MCNP while the Monaco/MAVRIC sequence of SCALE6.1 has been widely used in USA as a licensed code. The objective of this work is to perform a detailed comparative shielding analysis using the MCNP6 Monaco / MAVRIC sequence of and SCALE6.1 to show the suitability of these codes for KN-12.

2. Methods and Results

2.1 Computational method

In this work, a shielding analysis of KN-12 cask was performed using MCNP6 [2] and Monaco / MAVRIC module of SCALE 6.1 [3]. The MCNP6 code which was developed by LANL is a general-purpose, continuous-energy, generalized-geometry Monte Carlo radiation transport code designed to track many particle types and it has been widely used with variance reduction techniques for shielding calculations. On the other hand, the Monaco/MAVRIC sequence of SCALE 6.1 developed by ORNL provides a powerful FW-CADIS (Forward-Weighted-Consistent-Adjoint-Driven Importance-Sampling) methodology for variance reduction. The FW-CADIS methodology calculates the forward flux and adjoint flux using Denovo, three-dimensional S_N transport code, and automatically generates importance maps (i.e., weight windows) using MAVRIC. Finally, Monaco solves multi-group transport equation with Monte Carlo method using the previously generated importance map and variance reduction technique.

For MCNP6, the continuous energy library of ENDF / B-VIII.1 was used while 200 group neutron and 47 group gamma libraries were used in Monaco/MAVRIC sequence. The MCNP6 calculation was performed using geometry splitting as a variance reduction technique.

2.2 Shielding Analysis Procedure and Source Terms

The source term of the spent fuel shielding evaluation should be carefully considered. For gamma, the primary source term is mainly from the decay of the fission product. This primary gamma source terms are evaluated using ORIGEN-S with consideration of burnup and cooling time. In addition, the secondary gamma source terms should be evaluated to consider the contributions from the secondary photons released from neutron capture in fissionable and non-fissionable nuclides including structures. The secondary gamma transport calculation is performed using the neutron-gamma coupled transport calculations with a specified neutron source terms. In particular, the gamma source terms released from activation of the structures by fission neutrons under reactor operation should be prepared with ORIGEN-S. In this work, we considered only ^{60}Co activation which is most dominant. For neutron, the source terms are contributed from spontaneous fissions, (α,n) reactions, and the delayed neutrons produced by fission from subcritical multiplication, which are evaluated using ORIGEN-S.

We assumed that the spent fuel assemblies from Kori Unit 3 are loaded into KN-12. For These spent fuel assemblies have the discharge burnups ranging from 47,000 MWD/MTU to 49,500MWD/MTU and 4.50 uranium enrichment. In the shielding analysis, the composition of fuel was assumed to be fresh UO_2 fuel of 1.0wt% uranium enrichment, and the fuel assembly region and the upper and lower structure parts were homogenized. The neutron and gamma release rates of 12 loaded assemblies were estimated to be 9.564×10^8 neutrons/sec and 4.967×10^{16} photons/sec, respectively. The spectra of the neutron and gamma release rates are compared in Fig. 1. However, the gamma release rates given in Fig. 1 does not include the secondary gamma release rates such as the activation of ^{60}Co by fission neutrons.

The gamma release rates emitted by Co^{60} was calculated by multiplying the initial Co^{59} content of the structural material, the flux correction factor and the radioactivity per Co^{59} unit mass. The flux correction factor is used to consider axial flux distribution. Table 2 shows the gamma release rates contributed from activation of the structural sub-parts of 12 assemblies.

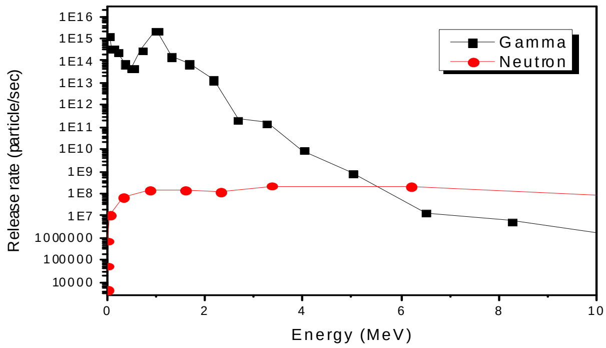
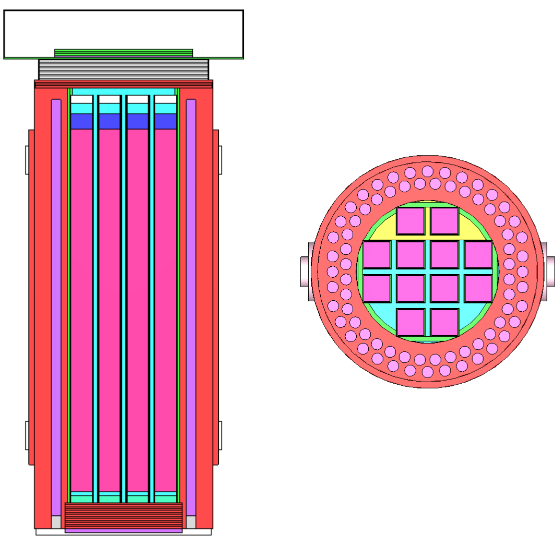


Fig. 1. Gamma and Neutron release rate of 12 assemblies

Table II: Gamma release rates originating from hardware activation products

Assembly sub-parts	Release rates (photons/sec)
Top end fitting	4.29×10^{13}
Plenum	5.66×10^{13}
Fuel region	5.15×10^{13}
Bottom end fitting	7.24×10^{13}

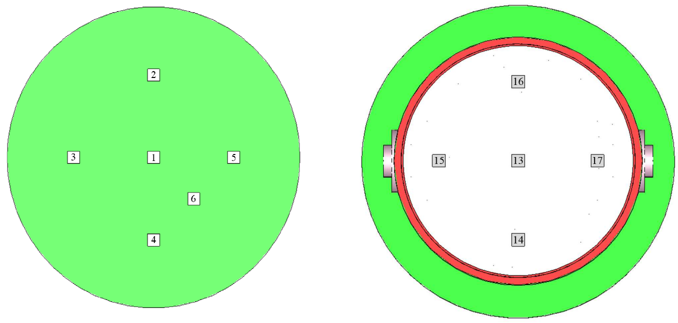
The axial burnup distribution is based on the conservative distribution of the WH type presented in the KN-12 safety analysis report [4]. It is assumed that the axial release rates of the gamma are proportional to the axial burnup distribution and that the axial release rates of the neutron are proportional to the fourth-order of the axial burnup distribution. We used the dose conversion factor recommended in ICRP-74 [5]. The cask modeled with SCALE6.1 is shown in Fig. 2.



(a) Axial (b) Radial

Fig. 2. KN-12 cask modeled with SCALE 6.1

The cask was modeled to contain 20% Helium and 80% water in a tip-over situation. We estimated the dose rates in the six tally regions on the upper impact limiter of the top part of the cask and in the five tally regions on the bottom lid. These tally regions are shown in Fig. 3.



(a) Upper impact limiter (b) Bottom lid

Fig. 3. Tally regions for dose rate estimation

2.3 Results

The gamma and neutron dose rates calculated by MCNP6 and SCALE6.1 at each tally region are compared in Tables III and IV, respectively. Table III shows that Monaco/MAVRIC under-estimates the gamma dose rates on the upper impact limiter by 11.0~23.6% than MCNP6. However, these discrepancies are considered to be acceptable and these dose rates are sufficiently less than the surface dose rate limit of 2mSv/hr. On the other hand, the discrepancies in the dose rates are much smaller on the bottom lid than the ones on the upper impact limiter (i.e., <10%). The highest dose rate was estimated in the tally region 13 of the bottom lid because it is in the central position.

Table III: Calculated gamma dose rates (μSv/hr)

Bottom lid				Upper impact limiter			
#	MCNP	SCALE	Diff (%).	#	MCNP	SCALE	Diff (%)
13	457.15	461.48	0.9	1	2.61	2.30	-11.8
14	123.80	120.05	-3.0	2	1.47	1.24	-15.2
15	173.76	158.47	-8.8	3	0.98	0.77	-21.1
16	130.16	119.22	8.4	4	0.85	0.65	-23.6
17	167.16	154.98	-7.3	5	0.96	0.86	-11.0
				6	1.41	1.21	-14.6

Monaco/MAVRIC also considerably under-estimates the neutron dose rates on the upper impact limiter than MCNP6 because the dose rates are quite lower than those on the bottom lid. It should be noted that the statistical errors of MCNP6 dose rates on the upper impact limiter were much larger than those of Monaco/MAVRIC. Similar to the gamma dose rates, the discrepancies in the neutron dose rates on the bottom lid is much less than those of the upper impact limiter (i.e., < 20.0%). These neutron dose rates are also sufficiently less than the surface dose rate limit of 2mSv/hr.

Table III: Calculated neutron dose rates(μSv/hr)

Bottom lid				**Upper impact limiter**			
#	MCNP	SCALE	Diff (%).	#	MCNP	SCALE	Diff (%)
13	105.42	96.44	-9.0	1	0.15	0.08	-46.5
14	55.42	55.19	-0.4	2	0.26	0.09	-65.3
15	67.82	70.64	4.0	3	0.14	0.02	-88.4
16	136.17	128.17	-6.0	4	0.09	0.02	-82.5
17	75.60	62.43	-17.0	5	0.13	0.04	-73.3
				6	0.14	0.03	-78.8

3. Conclusions

In this work, a detailed comparative shielding analysis for the KN-12 cask loaded with high burnup spent fuel assemblies from Kori Unit 3 was performed with MCNP6 and Monaco/MAVRIC. From the results, it was shown that these codes give the comparable surface dose rates and these dose rates are sufficiently lower than the surface dose rate limit. In particular, it is shown that the discrepancies of the surface dose rates on the bottom lid are less than 20% (<10% for neutron dose rate) while the larger discrepancies on the upper impact limiter are resulted from the large statistical errors due to its low dose rates.

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