Transactions of the Korean Nuclear Society Virtual spring Meeting
May 13-14, 2021

Comparison of SCALE 6.1 Monaco/MAVRIC and MCNP calculation for HI-STAR 63

Dong Jin Kim[a], Geon Hee Park[a], Ye seul Cho[c] and Ser Gi Hong[a]*, Ara Go[b], and Dae Sik Yook[b]

[a]Depart. of Nuclear Engineering, Hanyang University, 222, Wangsimni-ro, Seongdong-gu, Seoul, Republic of Korea
[b]Korea Institute of Nuclear Safety, 62 Gwahak-ro, Yuseong-gu, Daejon 305-338, Korea
[c]Korea Radioactive Waste Agency, 168, Gajeong-ro, Yuseong-Gu, Daejeon, Korea
*Corresponding author: hongsergi@hanyang.ac.kr

1. Introduction

Recently, storage and disposal of the SNF(Spent Nuclear Fuel) is a very serious issue in the South Korea. The SNFs from PWR's and CANDU's, each 20,053 and 474,176 SNF assemblies have been stored in temporary wet storage and dry storage as 2020 and it is expected that CANDU's SNF facilities will to be saturated in November 2021 without the additional expansion plan. Therefore, it will be clear that demands for transport and storage of PWR and CANDU increase near future.

Along with KINS, Hanyang University has developed the AMORES (Automatic Multi-batch Origin Runner for Evaluation of Spent Fuel) code automation system for automatically evaluating the source term of SNF and for automatically conduct criticality and shielding of cask. The purpose of this work was to verify and assess the HI-STAR 63 model to transport the CANDU SNF recently added to the shielding analysis module of the AMORES 4.0 through the comparisons with SCALE and MCNP.

2. Shielding Evaluation of the HI-STAR 63

1. SNF Assemblies Used to PHWR

In this work, we conducted the shielding analysis for HI-START 63 cask containing PHWR SNFs. The target fuel was CANDU 37 and natural uranium oxide was included to this fuel. It consists of 1, 6, 12, and 18 bundles from the center of the fuel, with an initial heavy metal of about 19.2 kg, which is small compared to 430 - 460 kg of PWR's SNF. The SNF assemblies discharged from the Wolseong nuclear power plant in Korea were transported and stored after cooling for at least six years, with an average burnup of 7,800 MWd/MTU per fuel, up to 12,000 MWd/MTU in South Korea.

2. Source Term

For PHWR, the source term was evaluated with two types of radiation sources. The first was neutrons from (α , n) reactions and spontaneous fission reactions. The second was the primary gamma-ray caused by the decay of fission products and actinides. The structural material activation considered by PWR was negligible because the SNF of PHWR did not use any of inconels or stainless steel containing Co-59 in the SNF assemblies. Also, the secondary gamma generated by a capture reaction of neutrons was not considered to PHWR SNFs because the neutron flux from PHWR SNFs was very smaller than in PWR SNFs.

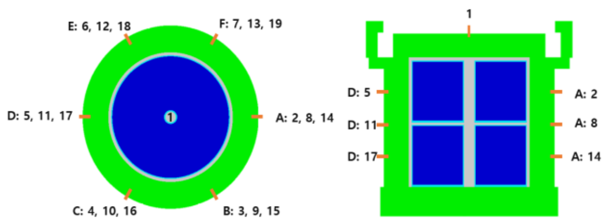
3. A Verification for the HI-STAR 63 of CANDU SNF Assemblies

This verification calculation was evaluated based on the results of the report [6]. The SNF used in the calculation was the SNFs having a cooling period of 8.3 years and 8.5 years after discharge. The verification calculations were conducted to different cases (i.e., casks 1 and 2) using different SNF parameters. The emission rates and total strength of radiation for each energy spectrum of the source were also based on the report [6]. The burnups, cooling periods, and specific powers applied are given in Table 1.

Table 1. SNF's specification for shielding evaluation

	HI-STAR 63, Cask 1		HI-STAR 63, Cask 2	
	Lower basket	Upper basket	Lower Basket	Upper basket
Burnup (MWd/MTU)	6,371.77-9,969.85	6,381.91-9,855.12	5686.19-9,678.17	5697.92-9641.34
Cooling time (y)	8.28-8.30	8.3	8.54-8.55	8.54-8.55
Specific power (MW/MTU)	22.55-52.47	22.37-53.85	15.27-47.68	20.48-51.02

The SCALE 6.1 ENDF v7-200n47g library was used for gamma and neutron calculation, with a batch of 10, particles of 1E+09 and 5E+07 per batch. Tallies were evaluated at the points 2m away from the surface of the HI-STAR container and the points just on the container surface. As shown in Fig. 2, the tallies were evaluated radially at the six positions and axially three positions for each radial position. Also, in addition, one tally was given to the center of the top surface.



.

Fig. 2. HI-STAR 63 model for shielding

The dose rates at the tally positions evaluated using SCALE 6.1 Monaco/MAVRIC codes ware compared with those obtained using MCNP6 for both casks. In the cask 1, the largest discrepancy of -7.71% on the surface occurred at the point 6, and the RMS was 3.94% at the total tally mesh. At the positions 2m away from the surface, the largest discrepancy of 5.03%, was found at the point 6. The RMS of 3.43% at the total tally mesh.

For the cask 2 surface, the largest discrepancy was 5.15% at the point 19 and the RMS of -0.2% at the total tally mesh. At the positions 2m away from the surface, the largest discrepancy of 28.86% occurred at the point 3 and the RMS of -0.28% at the tally mesh. Except for a cask2 at 2m away from the surface, all values are within the standard deviation. The dose error rate for cask2 at 2m away from the surface was derived from four locations of MCNP6 mesh tallies. This error can be solved by increasing the number of particles. Also, it is considered that these discrepancies in dose rate may be caused by the differences in the cross-section libraries (continuous and multi-group libraries in MCNP6 and SCALE 6.1 Monaco/MAVRIC codes, respectively).

Table 2. Comparison of the total dose rates obtained with MCNP and SCALE 6.1 (Maximum discrepancies and the positions where they occur)

Cask1 (calculated by SCALE 6.1)					
	Position	**Point**	**Total dose rate(μSv/h)**	**Error rate (%)**	**RMS (%)**
Surf Max	E	6	283.84	-7.71	3.94
2 m Max	E	6	27.12	5.41	3.43

Cask2 (calculated by SCALE 6.1)					
	Position	**Point**	**Total dose rate(μSv/h)**	**Error rate (%)**	**RMS (%)**
Surf Max	F	19	233.75	5.15	4.10
2 m Max	B	3	24.68	28.86	10.31

3. Conclusions

The purpose of this work was to verify the shielding model used in AMORES4.0 for the HI-STAR 63 carrier using SCALE 6.1 based on MCNP6 geometry input used in AMORES4.0's report [6]. This verification was performed by comparing the results of MCNP6 calculations given in the report [6] and the SCALE 6.1 code ones used in AMORES4.0. The verifications were done for two types of casks. For the type 1 cask, the largest discrepancies of the total dose rate on the surface and 2m away from the surface were -7.71% and 5.41%, respectively. For the type 2, the largest discrepancies of the total dose rate on the surface and 2m away from the surface were 5.15% and 28.86%, respectively. Also, the total dose rates are shown to be much was far below 2 mSv/hr surface limit and 10 mSv/hr exclusive-use shipments. These levels of discrepancy can be reasonable in the shielding design if the calculated dose rates were all much lower than the dose limits.

REFERENCES

[1] HOLTEC INTERNATIONAL, HI-STAR 63 Type B(U) Transport System for CANDU Spent Fuel
[2] S. G. HONG et al., Development of a Code for Regulation and Verification of Back-End Fuel Cycle Facilities, KINS/HR-1613, Korea Institute of Nuclear Safety. 2017
[3] I. C. Gauld, O.W. Hermann, R. M. Westfall, ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Build and Decay, and Associated Radiation Source Terms, ORNL/TM-2005/39 Version 6.1, Section F7, Oak Ridge National Laboratory, 2009.
[4] D. E. PELPLOW, S. M. BOWMAN, James E. HORWEDEL, J. C. WAGNER, “Monaco/MAVRIC: Computational Resources for Radiation Protection and Shielding
[5] J.C. Wagner, C.V. Parks, Recommendations On The Credit For Cooling Time In PWR Burnup Credit Analyses, NUREG/CR-6781 (ORNL/TM-2001/272), Oak Ridge National Laboratory, 2003.
[6]Yeonghwan Choi, 중수로 사용후핵연료 운반용기 사용검사 차폐 성능해석 보고서, 2019.12