Transactions of the Korean Nuclear Society Autumn Meeting
Yeosu, Korea, October 25-26, 2018

Application of MCS Code for Spent Fuel Cask Analysis

Mai Nguyen Trong Nhan, Chidong Kong, Hyunsuk Lee, Sooyoung Choi, Deokjung Lee*
Department of Nuclear Engineering, Ulsan National Institute of Science and Technology, 50 UNIST-gil, Ulsan, 44919, Republic of Korea
**Corresponding deokjung@unist.ac.kr*

1. Introduction

Monte Carlo code MCS developed in Ulsan National Institute of Science and Technology (UNIST) is applied and verified to spent fuel cask analysis. After 40 years of operation. the Kori unit 1 was shutdown for decommissioning. On average, one third of the fuel assemblies are replaced after each cycle, resulting in nearly hundreds of spent fuel assemblies are resting in the spent fuel pool of Kori unit 1. The spent fuel located inside the cooling pool must be transported out of the site to enable the decommission process. Thus, there is a demand for spent fuel casks capable of keeping spent fuels as well as personnel safe during handling and transportation of the spent nuclear fuel. In this study, criticality of various spent fuel casks, namely KN12, KN18, GBC32, KORAD21 are analyzed together with the flux calculation for KORAD21, using the Monte Carlo code named MCS developed in UNIST. The calculated results are also compared with results from MCNP6 and KENO VI code.

2. Methods and Results

2.1 MCS

MCS is a 3D continuous-energy neutron-physics code for particle transport based on the Monte Carlo method, under development at UNIST since 2013 [1, 2]. Two kinds of calculations are allowed by MCS: criticality runs for reactivity calculations and fixed-source runs for shielding problems. MCS neutron transport capability is verified and validated with many benchmark problems including BEAVERS benchmarks, the International Criticality Safety Benchmark Experimental Problem (ICSBEP) and Jordan Research and Training Reactor (JRTR).

2.2 Simulation

Casks are modeled in MCS, MCNP6 and KENO VI. All simulation used ENDF/B-VII.1 library. Geometry details for KN12, KN18, GBC32 and KORAD21 are provided in document [3,4,5,6] The criticality analysis is carried out with two type of fuel composition: fresh and burned fuel. In case of fresh fuel, the fuel assembly is modeled as a 17x17 fuel assembly with an enrichment of 3.5 wt.%. For the burned fuel composition, STREAM code (Steady state and Transient REactor Analysis code with Method of Characteristics [7], also developed in UNIST) is used to generate a composition of a 50MWd/kg 17x17 fuel assembly with 1481 Effective Full Power Day. The source terms for this burned fuel composition are also calculated by STREAM. Burned fuel composition and source terms are fed into MCS and MCNP for criticality and fixed source simulation.

2.3 Result for criticality

Each criticality simulation is executed with 50 inactive cycles, 500 active cycles and 30,000 histories per cycle. k_{eff} for fresh fuel and burned fuel are presented in Table I and Table II, respectively.

Table I: k_{eff} for burned fuel

Cask	Code	k_{eff}	Diff. vs. MCNP (pcm)
KN12	MCNP6	0.94749±0.00019	Ref.
	MCS	0.94722±0.00020	-27
	KENO VI	0.94715±0.00022	-34
KN18	MCNP6	0.97197±0.00020	Ref.
	MCS	0.97187±0.00022	-10
	KENO VI	0.97166±0.00020	-31
GBC32	MCNP6	1.06637±0.00019	Ref.
	MCS	1.06687±0.00020	50
	KENO VI	1.06697±0.00021	60
KORAD21	MCNP6	0.91653±0.00020	Ref.
	MCS	0.91667±0.00022	14
	KENO VI	0.91660±0.00020	7

Table II: k_{eff} for burned fuel

Cask	Code	k_{eff}	Diff. vs. MCNP (pcm)
KN12	MCNP6	0.61374±0.00014	Ref.
	MCS	0.61412±0.00016	38
KN18	MCNP6	0.56614±0.00014	Ref.
	MCS	0.56592±0.00015	-22
GBC32	MCNP6	0.69636±0.00014	Ref.
	MCS	0.69654±0.00016	18
KORAD21	MCNP6	0.54654±0.00015	Ref.
	MCS	0.54676±0.00015	22

The k_{eff} results show good agreement between MCS, MCNP6 and KENO VI. Differences between MCS and MCNP6 is less than 60 pcm and the differences between MCS and KENO is less than 20 pcm.

2.4 Result for fixed source simulation

Among the abovementioned casks, KORAD21 is chosen for neutron and photon flux calculation. Mesh tally for flux calculation is applied at the horizontal mid plane and vertical mid plane of the cask as in Fig. 1. The mesh size is 10x10x10 cm^3 and 200 million histories are used.

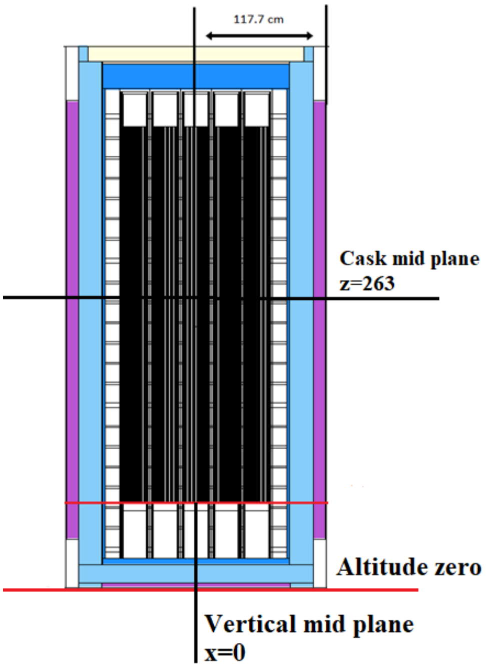

Fig. 1. Tallied position for KORAD 21 cask.

The neutron flux at the horizontal mid plane and vertical mid plane of the cask are shown in Fig.2 and Fig.3, respectively.

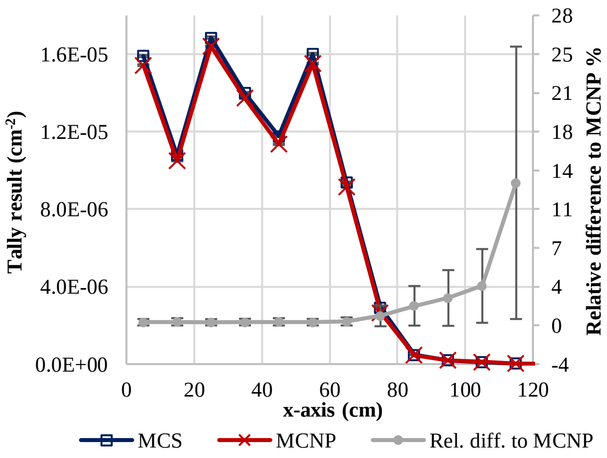

Fig. 2. Tally result for neutron at x=0 (cask center) to 120 (outside cask), y=0, z=263

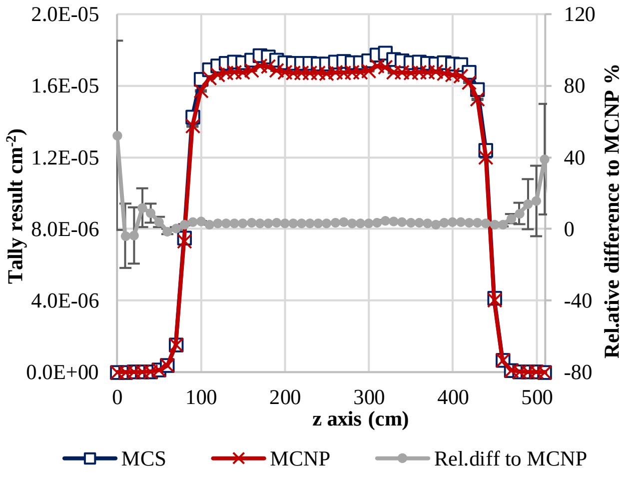

Fig. 3. Tally result for neutron at x=0, y=0, z=0 (cask bottom) to 528 (cask top)

The photon flux at the horizontal mid plane and vertical mid plane of the cask are shown in Fig.4 and Fig.5, respectively.

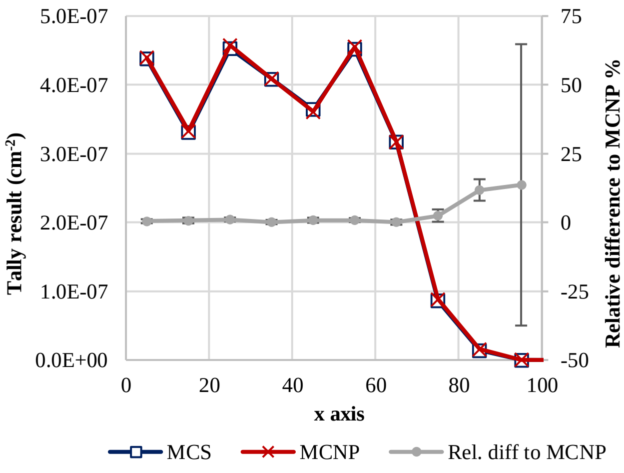

Fig. 4. Tally result for photon at x=0 (cask center) to 95, y=0, z=263

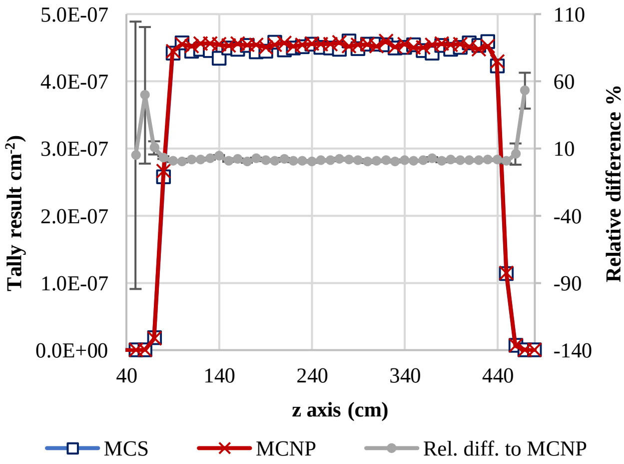

Fig. 5. Tally result for photon at x=0, y=0, z=40 to 490

The positions of peaks in Fig.2 and Fig. 4 represent the centers of fuel assemblies. The photon flux is much lower compared to the neutron flux MCS (~100 times). Results from MCS and MCNP6 are relatively consistent. within the cask inner space. On the horizontal mid plane, the relative difference of MCS to MCNP6 in neutron flux and photon flux values is less than 10% for x<115 cm and x<80 cm, respectively. On the vertical mid plane, relative difference for neutron flux is less than 10% within range of z from 30 to 480 cm and for photon is from 60 to 460 cm. Outside the abovementioned x and z range (positions close to the periphery of the cask where shielding materials are located and region outside of the cask), neutron flux and photon flux have large error due to very low number of non-zero scores in the tallied zone. Flux values in these positions are unreliable and difficult to calculate without using any variance reduction technique. Small difference between MCS and MCNP6 within the cask inner space can be due to different type of tally used in the problem (track length tally is used in MCNP6 and collision tally is used in MCS).

3. Conclusions

In this study, the application of MCS code for spent fuel cask analysis has been presented. The multiplication factor obtained from MCS shown good agreement with results from other Monte Carlo code. Large relative error in flux values are observed at periphery positions of shielding materials and outside of the cask. Variance reduction technique is, therefore, necessary to obtain reliable result in flux as well as in dose rate calculation. Weight window would be applied in MCS and flux calculation using this variance reduction technique will be study further.

Acknowledgement

This research was supported by the project(L18S015000) by Korea Hydro & Nuclear Power Co. Ltd.

REFERENCES

[1] J. Jang, W. Kim, S. Jeong, et al., "Validation of UNIST Monte Carlo code MCS for criticality safety analysis of PWR spent fuel pool and storage cask," *Annals of Nuclear Energy*, Vol. 114, 2018, pp.495-509.

[2] H. Lee, W. Kim, P. Zhang, et al., "Preliminary simulation results of BEAVRS three-dimensional Cycle 1 whole core depletion by UNIST Monte Carlo code MCS," Proceedings of the M&C conference, Jeju, Korean Nuclear Society, 2017 Apr 16-20. [USB]

[3] S. H Chung, J. H. Yoon, K. H Yang, J. M. Kim
H. Y. Jee, J. H. Ha and M. J. Song, "Evaluation of the KN-12 Spent Fuel Shipping Cask", Proceedings of the Korean Nuclear Society Spring Meeting, Cheju, Korea, May 2001

[4] Safety Test Report for the KSNPR(KN-18) Transport Cask. KAERI/TR-3659/2008

[5] J. C. Wagner, Computational Benchmark for Estimation of Reactivity Margin from Fission Products and Minor Actinides in PWR Burnup Credit. Oak Ridge National Laboratory. NRC Job Code W6479.

[6] W. S. Choi S. S. Cho, K. S. Seo, Y. Y. Yang, S. W. Yu. "A Study on Puncture Simulation of Transportation Cask". Transactions of the Korean Nuclear Society Spring Meeting, Jeju, Korea, May 17-18, 2018.

[7] S. Choi, C. Lee, D. Lee, "Resonance Treatment using Pin-Based Pointwise Energy Slowing-Down Method," J. Comput. Phys., 330: 134-155.