2017 Water Reactor Fuel Performance Meeting
September 10 (Sun) ~ 14 (Thu), 2017
Ramada Plaza Jeju • Jeju Island, Korea

SENSITIVITY CALCULATIONS FOR CRITICALITY AND SHIELDING ANALYSES USING SAMPLER

Kyoon-Ho CHA[1], Min-Chul KIM[2], Do-Yeon KIM[3], Seo-Jeong LEE[4], and Tai-Wan KIM[5]

Korea Hydro & Nuclear Power Co., Ltd. 1312, Yuseong-daero, Yuseong-gu, Daejeon, Korea, 34101,
khcha.cri@khnp.co.kr[1], kim.minchul@khnp.co.kr[2], doyeon.kim@khnp.co.kr[3], s.jeong2@khnp.co.kr[4]
Hanyang University, 222, Wangsimni-ro, Seongdong-gu, Seoul, Korea, 04763, taiwan@hanyang.ac.kr[5]

ABSTRACT: *Sensitivity calculations for criticality safety analysis of a spent fuel storage and shielding analysis of a storage cask were performed using SAMPLER sequence in SCALE6.2.1. STARBUCS and MAVRIC sequence in SCALE were also used for the criticality and shielding analysis. Two kinds of sensitivity models for criticality analysis were made for a spent fuel storage pool using SAMPLER. The criticality calculations were performed for searching both the optimum density of water due to the change of the temperature of the spent fuel pool and the optimum moderator axial thickness for the criticality calculation model of spent fuel storage rack. Also, another two sensitivity models for shielding analysis were made for a storage cask with 24 PWR spent fuel assemblies using SAMPLER. The gamma dose-rates were calculated by changing the thickness of gamma shield of the cask and calculated on the cask surface from bottom to top of the cask.*

KEYWORDS: *Criticality, Shielding, SAMPLER, STARBUCS, MAVRIC, SCALE6.2.1*

I. INTRODUCTION

Sensitivity calculations for a criticality and shielding analyses were performed using SAMPLER [1]. SAMPLER will be available for the first time in SCALE 6.2. SAMPLER is a “super-sequence” that performs general uncertainty analysis for SCALE6.2 sequences by statistically sampling the input data and analyzing the output distributions for specified responses. SAMPLER samples the input randomly and perturbed a wide range of parameters and nuclear data within the SCALE code package, that is, the user input varies considerably based on the type of perturbations being applied. SAMPLER generates a specified number of perturbed inputs for each case. Identical perturbed values are used in each realization of each case, as specified. Different values are used for cases that are not specified to use the same values. SAMPLER uses a three-step process for executing the required calculations. The first step is the generation of the perturbed inputs. The second step is to execute all the generated SCALE calculations. After all calculations are complete, SAMPLER is run in a post-processing mode to extract requested information from the generated output.

Criticality calculations with burn-up credit of spent fuel racks for several plants using STARBUCS [2] module in SCALE6.2.1 were performed for an accident related to the loss of cooling water in the spent fuel pool due to unexpected conditions. Considering the unexpected conditions, the most critical condition will be happened for the situation that is the most optimal moderated condition in the pool. These criticality calculations according to the change of the moderator density of temperature-dependent could be performed using SAMPLER. Another sensitivity calculations of criticality model for an axial moderator thickness change of a spent fuel storage rack were also performed using SAMPLER.

Shielding analyses were also performed using MAVRIC [3] sequence in SCALE6.2.1. MAVRIC sequence uses multi-group shielding code MONACO [4] and calculates the adjoint flux as a function of position and energy. The multidimensional cask calculations in this study were primarily performed with MAVRIC sequence utilizing the SCALE 200-neutron/47-photon-group cross section based on ENDF B-VII.0 libraries and ANSI/ANS 6.1.1-1977 flux-to-dose-rate factors [5]. Sensitivity calculations for the Westinghouse MC-10 forged-steel storage cask with 24 PWR spent fuel assemblies were performed for two kinds of sensitivity models using SAMPLER with MAVRIC. [6] Gamma dose-rates were calculated by changing the thickness of the gamma shield, and by changing the height of the cask surface from bottom to top of the cask.

II. EVALUATIONS AND RESULTS

II.A. Criticality Analysis of Spent Fuel Pool

Criticality analyses were performed to find the optimum moderator density of spent fuel pool and optimal axial moderator thickness to model a spent fuel storage rack. For the spent fuel pool, sensitivity evaluations of criticality analysis were done to see the trends of the criticalities of various storage racks using SAMPLER. Criticality calculations with burn-up credit of various spent fuel racks using STARBUCS were performed for an accident related to the loss of cooling water in the spent fuel pool due to unexpected conditions. These criticality calculations according to the change of the moderator density of temperature-dependent were to be performed using SAMPLER and STARBUCS. The nuclides considered for application of burnup credit were 28 nuclides (major actinides and fission product) recommended by ISG 8 (Rev.3) [7] for conservative assumption. Fig. 1 shows the results of the trends of the criticalities of various racks for the change of the moderator density.

Another sensitivity calculations of criticality model for an axial moderator thickness change of a spent fuel storage rack were also performed using SAMPLER, STARBUCS and ISG 8 (Rev.3). Fig. 2 shows the results of the trends of the criticalities for k-eff verse axial thickness of h2o for spent fuel pool.

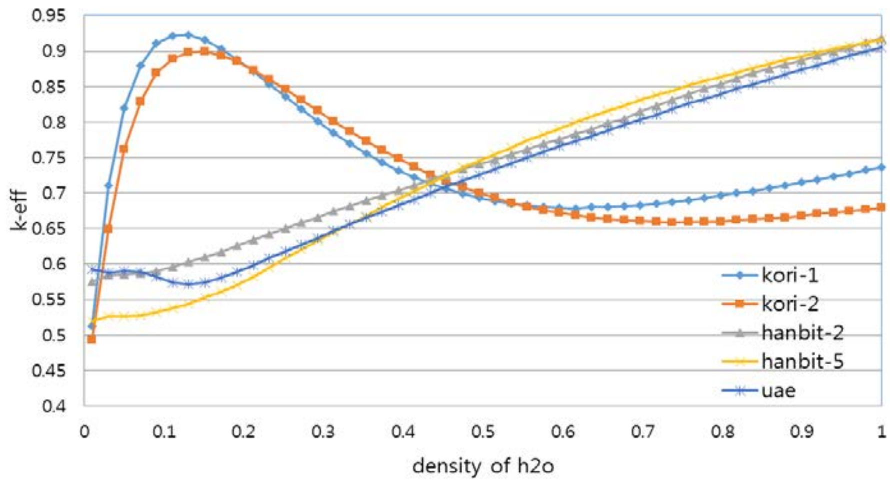

Fig. 1. Trends of the criticalities for various spent fuel storage racks

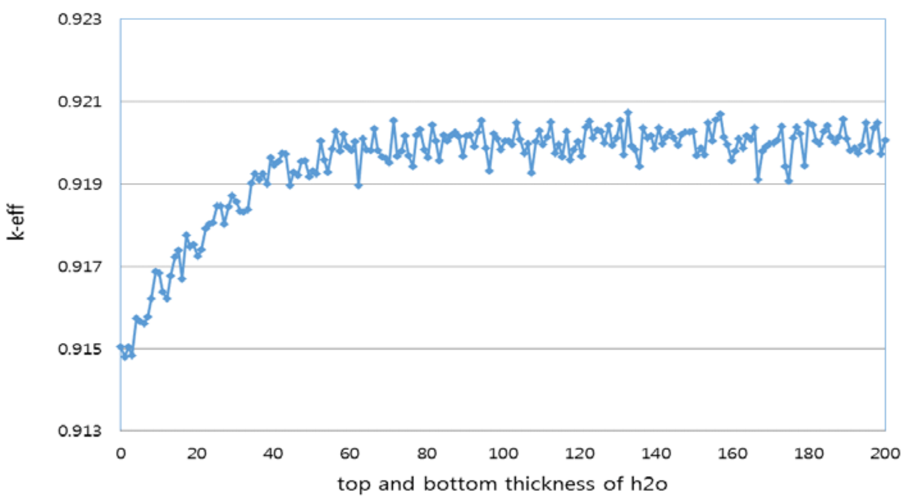

Fig. 2. Trends of the criticalities for k-eff verse axial thickness of h2o for spent fuel pool

II.B. Shielding Analysis of Spent Fuel Storage Cask

Shielding analysis model is made with the Westinghouse MC-10 forged-steel storage cask. The MC-10 PWR spent fuel storage cask consists of a low-alloy forged-steel body, and is 4.8m long and 2.7m in diameter. The fuel basket within the cask is configured to hold 24 PWR spent fuel assemblies and is constructed of aluminum. The spent fuel assemblies used in this modeling are of a standard WH 15x15 rod design. For gamma dose evaluations, three kinds of fuel assemblies are loaded in the cask. MAVRIC sequence in SCALE6.2.1 uses multi-group shielding code MONACO and calculates the adjoint flux as a function of position and energy, and combines the results of an adjoint calculation from the 3-D deterministic code TORT with MONACO which is a new 3-D Monte Carlo code being developed within SCALE code package for shielding calculations. The cask shielding calculations were primarily performed with the SCALE 200-neutron and 47-photon-group cross section based on ENDF/B-VII.1 libraries and ANSI/ANS 6.1.1-1977 fluence-to-dose factors. The sensitivity evaluations of gamma responses with gamma shield were performed to use the new SAMPLER sequence within the release of the newest version of SCALE code package.

Sensitivity analyses were performed by changing the thickness of the gamma shield and axial position of the cask. SAMPLER calculation provides the expected values of the data and repeats the perturbation for a specified number of samples (set by the user) to obtain the results distribution with its standard deviation and its correlation coefficients. Each SAMPLER module has been used coupled with MAVRIC.

First, when the thickness of gamma shield was changed, the other structures which are the neutron shield, fins and boundaries were to be changed. The design thickness of the gamma shield was 25.7cm and was changed from 15.7cm to 35.7cm. It was discretized into 60 shields for sensitivity simulations by changing the thickness of the shield. Fig. 3 shows the results of the simulations using SAMPLER. The point of the design thickness of MC-10 cask is 112.4 on the x-axis of the graph. This means that the thickness of gamma shield was designed appropriately.

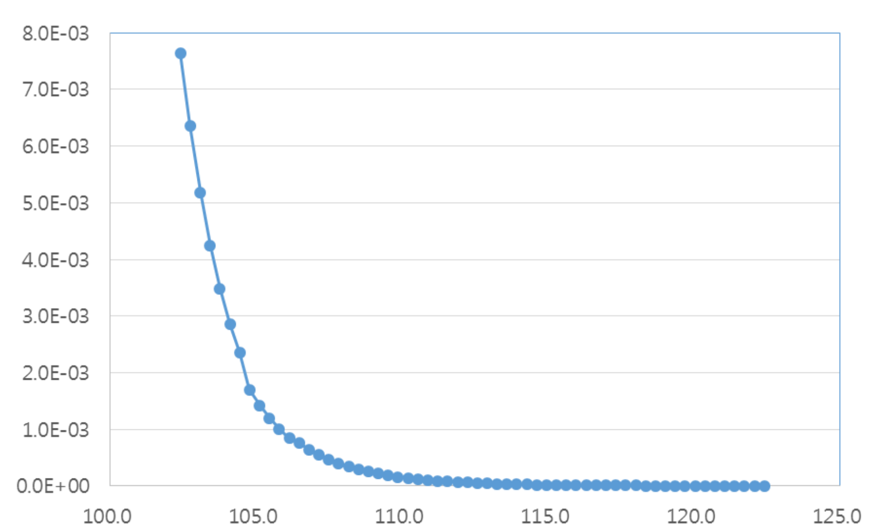

Fig. 3. Trends of the gamma responses for the change of gamma shield thickness

Next, the gamma responses for the axial positions of the cask were calculated. The height of the cask is 473.6cm (-225.6~248.0cm). The number of positions to be calculated was 60 points varied from bottom (-225.6cm) to top (248.0cm) of the cask. The gamma dose-rates were calculated on every 60 surfaces from bottom to top of the cask using SAMPLER and MAVTIC. Fig. 4 shows the responses according to the positions of the cask height.

The peaks of responses on bottom and top might come from the absent of cooling fins made with carbon steel. And the drop of the response on the top might be stemmed from the reason that there is a space between a test lid on the top of the cask and spent fuel assemblies in the cask.

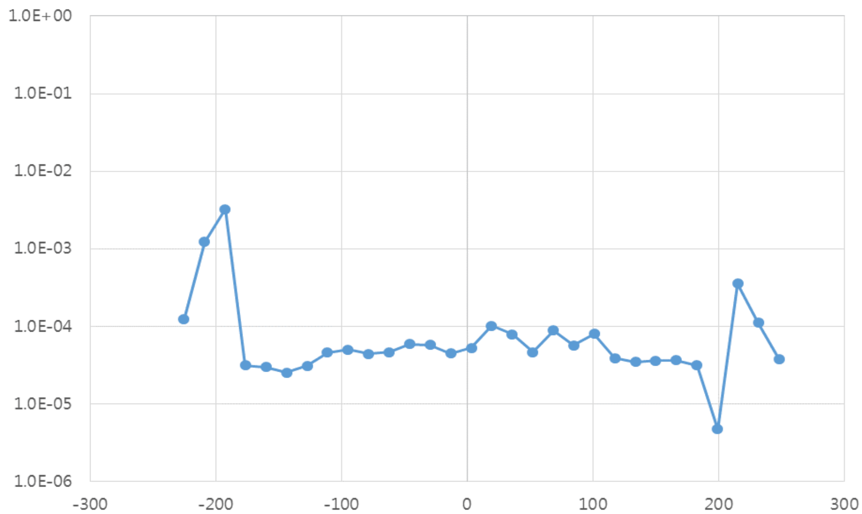

Fig. 4. Trends of the gamma responses for the change of axial heights of the cask

III. CONCLUSIONS

Sensitivity calculations for criticality analysis of a spent fuel storage pool and shielding analysis of a storage cask were performed with STARBUCS, MAVRIC which uses MONACO Monte Carlo code for shielding calculations, and SAMPLER. Based on the results of those calculations, sensitivity evaluation models of both criticality calculations using SAMPLER with STARBUCS and shielding calculations of gamma shield using SAMPLER with MAVRIC were discovered to be determined properly. Using those sensitivity models of criticality and shielding analyses, various parametric studies could be performed more efficiently and correctly than in the past.

Furthermore, additional study for the shielding analysis is needed in response analysis of neutron shield. After these study, an optimum shielding calculation modeling for the determination of design specifications of the gamma and neutron shield for a future storage and transport cask design, if needed, will be established.

ACKNOWLEDGMENTS

This study was supported by its own budget of Korea Hydro & Nuclear Power Co., LTD.

REFERENCES

1. M. L. Williams, et al., “SAMPLER: A Module for Statistical Uncertainty Analysis with SCALE6.2 Sequences,” 2016.
2. G. Radulescu & I.C. Glaud, “STARBUCS: A SCALE Control Module for Automated Criticality Safety Analyses Using Burnup Credit,” 2016.
3. D.E. Peplow and C. Celik, “MAVRIC: MONACO with Automated Variance Reduction using Importance Calculations in SCALE6.2.1,” 2016.
4. D.E. Peplow and C. Celik, “MONACO: A Fixed-Source Monte Carlo Transport Code for Shielding Applications in SCALE6.2,” 2016.
5. ANSI/ANS 6.1.1-1977, “Neutron and Gamma-ray fluence-to-dose Factors”
6. EPRI, TR-104329, “Evaluation of Shielding Analysis Methods in Spent Fuel Cask Environments,” 1995.
7. ISG-8 Rev.3, “Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks,” U.S. NRC, September 18, 2012.