Effects of Neutron Absorbing Nuclide on Criticality of Spent Nuclear Fuel Storage Systems

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1. Introduction

Recently, the capacity of the current spent nuclear fuel storage system (wet storage) in South Korea is almost saturated by discharged spent fuel. [1]. Dry storage of spent nucleaer fuel is one of the solutions to the capacity shortage problem of the wet storage system. Dry storage methods have advantages in expandability, transportability, simplicity, and passive safety[2, 3]. Therefore, there are many on-going studies on the development of efficient dry storage systems around the world.

A basket is one of the critical components of a dry storage system. The primary function of the basket is neutron absorption, in addition to heat removal and structural support [4–6]. Therefore, an understanding of the effects of neutron absorbing nuclides is essential for the development of advanced basket materials.

In this study, the role of the neutron absorbing nuclides in a basket is investigated by Monte Carlo simulation with MCNP. A simulation was conducted by various geometry, nuclides, and concentration. Real scale simulation is analyzed by a simplified geometry and ideal conditions. The performance of Gd, Cd and B were investigated. The result of this study might provide useful guidance for developing next-generation spent nuclear fuel storage systems.

2. Methods and Results

2.1 Simulation Condition of MCNP Code

GBC 32 cask and simplified core-shell shape geometry were prepared for the simulation (Fig. 1). The purpose of GBC 32 cask simulation was to figure out an actual performance of neutron absorbing nuclide in the actual scale of spent fuel storage cask and the neutron absorbing characteristics were investigated by simplified the core-shell geometry simulation. Water, vacuum, and helium were adopted as moderators and the B, Gd, Dy, Eu, Sm, Cd, and Er were adopted as neutron absorbing nuclides. The criticality was calculated by K-code of MCNP simulation. K-code was used to get k-value. The total numbers of the cycle of simulation were 800 and first 200 cycles were skipped. 1000 histories per cycles were used to ensure error lower than 0.1%

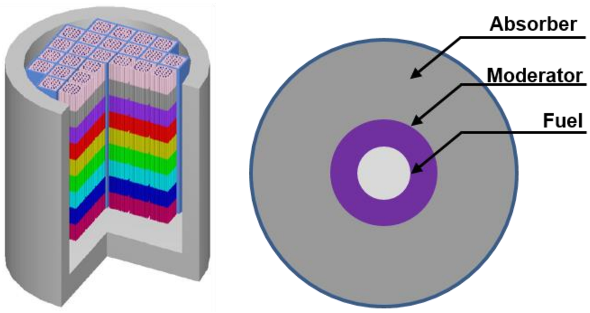

Fig. 1. The geometry of GBC 32 cask (left) and simplified core-shell structure (right).

Fig. 2 shows the criticality of GBC 32 cask when nuclides Gd, B, and Cd were used as neutron absorbing nuclides. Gd shows the lowest criticality margin while it has the highest neutron absorbing cross-section.

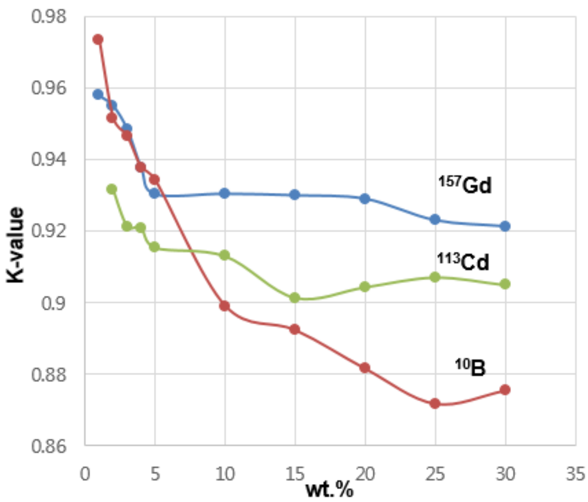

Fig. 2. Criticality of the GBC32 cask with various neutron absorber

2.2 Contribution to Criticality of Various Nuclides

In the simplified core-shell geometry, various neutron absorbing nuclides, B, Gd, Dy, Eu, Sm, Cd, and Er, were input in various condition to find out the neutron absorbing characteristics of each nuclide. The simulation was conducted in equi-atomic number density, equi-macroscopic neutron scattering cross-section, and equi-macroscopic neutron absorption cross-section. Two types of neutron energy levels were simulated, fast and fully moderated.

The result of simplified core-shell geometry simulation was listed in table 1. Even though ^{157}Gd has 20 times larger neutron absorbing cross-section, but ^{10}B show lower criticality when neutron is moderated. In case of fast neutron, only slight improvement of ^{157}Gd is observed.

Table I: Criticality of simplified core-shell geometry

	^{157}Gd	*^{10}B*
Macroscopic CX (cm^{-1})	*7573*	*580*
Water	*0.54403*	*0.53879*
He	*0.35161*	*0.35138*
Vacuum	*0.34879*	*0.35095*

2.3 Criticality and Concentration of Nuclide

Fig. 3 show the absorbing nuclide concentration effect on the criticality of a cask. In fig.3, Gd show better criticality in low concentration level (below 2 vol.%) and B show better criticality when high concentration level (above 10 vol.%)

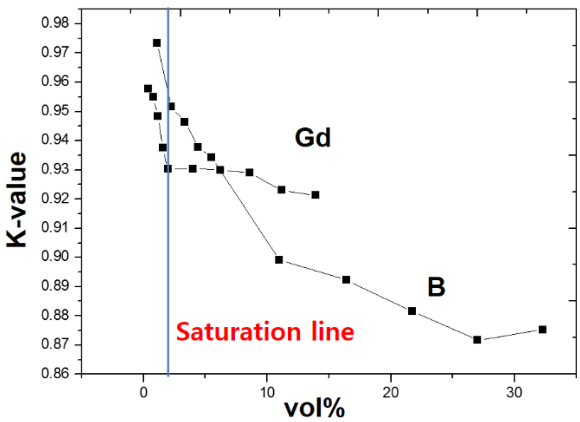

Fig. 3. Effect of nuclides concentration on the criticality

3. Conclusion

In this study, the actual cask condition simulation and ideal core-shell structure cask simulation were conducted with various neutron absorbing nuclide. Interestingly, Gd showed the lowest performances while it has the highest neutron absorption cross-section. Furthermore, the Performance of B is better than Cd and Gd despite its small neutron absorption cross-section.

In order to investigate the phenomena, the simplified core-shell cask simulation was conducted with various nuclides and their isotopes. The result shows the stronger relationship between criticality and ratio of scattering and absorption cross-section than criticality and absorption cross-section. Therefore, B is better than Gd due to its low scattering cross-section to absorption cross-section ratio.

The criticality control of storage system can be distinguished as two stages by concentration of absorption nuclides, first small total cross-section area stage and large cross-section area stage. In small cross-section area stage the number of the fate of generated neutron is three, penetrating absorber nuclides, being absorbed by absorber nuclides, and being scattered by absorber nuclides. Therefore, the larger absorption cross-section is better due to the lower number of penetration. However, in large cross-section area level, penetration of neutron occurs no long, so a ratio of scattering to absorption becomes more important.

This study might accelerate the developing the next generation spent fuel storage cast and its optimization.

ACKNOWLEDGMENT

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REFERENCES

[1] H. Dho, T. Kim, and C. Cho, The Evaluation of Minimum Cooling Period for Loading of PWR Spent Nuclear Fuel of a Dual Purpose Metal Cask, vol. 14, no. 4, p. 411–422, 2016.
[2] R. R. DeVoe, K. R. Robb, and S. E. Skutnik, Sensitivity analysis for best-estimate thermal models of vertical dry cask storage systems, *Nucl. Eng. Des.*, vol. 320, p. 282–297, 2017.
[3] M. J. Leotlela, Criticality Safety Analysis of the Design of Spent Fuel Cask , Its Manipulation and Placement in a Long-Term Storage, University of the Witwatersrand, Ph.D. Thesis, 2015.
[4] J. Kusui, K. Hayashi, K. IWASA, and M. IWASE, Development of Basket for Transport/Storage Cask using Square Tube made of Aluminium Alloy containing Neutron Absorbing Materials, Transportation of Radioactive Materials (PATRAM 2004), September 20-24, 2004, Berlin, Germany
[5] M. H. Industries, Development on Spent Nuclear Fuel Transport and Storage Cask,Mitsubishi Heavy Industries Technical Review, vol. 43, no. 4, p. 1–6, 2006.
[6] J. B. Wierschke and L. Wang, Evaluation of Aluminum-Boron Carbide Neutron Absorbing Materials for Interim Storage of Used Nuclear Fuel, NEUP project 10-603, p. 1–159, 2015.