Proceedings of the 19th International Symposium on the
Packaging and Transportation of Radioactive Materials
PATRAM 2019
August 4-9, 2019, New Orleans, LA, USA

EVALUATION OF NEUTRON FLUX OF A PWR DRY STORAGE CASK

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ABSTRACT

In Korea, twenty-four nuclear power plants are currently under operation. As the capacity of spent fuel storage facilities becomes saturated, a dry storage facility will be constructed. To verify the safety of spent fuels in a dry storage facility, several studies have been conducted. Most previous studies have proposed a technique based on the detection of gamma-rays. However, since gamma-rays may be emitted from sources other than spent fuel, the detection of gamma-rays have uncertainty. To reduce uncertainty, we developed an evaluation technique based on the neutron detection. The objective of this study was to evaluate neutron flux on the surface of a PWR dry storage cask. TN-32 dry storage cask was selected for this evaluation study. TN-32 dry storage cask is a canister-based cask which was developed by AREVA-TN. Neutron flux on the surface of a dry storage cask was evaluated using computer codes. To define source term, ORIGEN-APR code was used to calculate neutron emission rate and energy spectra from a WH 17 x 17 type spent fuel. Axial distribution of neutron emission rate from a fuel rod was calculated considering axial burnup distribution. MCNP code was used to simulate geometry and material properties of spent fuel assemblies and a dry storage cask. Neutron fluxes for TN-32 were calculated on the surface of top, side and bottom of a cask after 10 years of cooling time. For TN-32 dry storage cask, average neutron fluxes were evaluated to be 1.08×10^3 neutrons/cm^2·sec, 4.50×10^2 neutrons/cm^2·sec, and 1.04×10^3 neutrons/cm^2·sec on the surface of top, side, and bottom of a cask, respectively. Axial neutron flux distributions on the side part varied with cask structure. The reason for high neutron fluxes on the surface of top and bottom is that neutron shields were designed to have little effect on the top and bottom. This study can be applied to evaluate the characteristics of spent fuels and neutron flux of a dry storage cask. Furthermore, this study can be conducted with measurement data from neutron flux of a dry storage cask.

INTRODUCTION

There are twenty-four commercial nuclear power plants in Korea. Spent fuels generated during operation have been stored in the temporary storage facilities on the site of nuclear power plants. However, the capacity of the temporary storage facilities is expected to be saturated in 2024. To solve this problem, the Korea government is planning to construct a dry storage facility by 2035 [1]. To verify the safety of spent fuels in a dry storage facility, several studies have been conducted. Most previous studies have proposed a technique based on the detection of gamma-rays. However, since gamma-rays may be emitted from sources other than spent fuel, the detection of gamma-rays have uncertainty [2]. To reduce uncertainty, we developed an evaluation technique based on the neutron detection. The objective of this study was to evaluate neutron flux on the surface of a PWR dry storage cask.

MATERIALS AND METHODS

ORNL isotope generation and depletion-Automatic Rapid Processing (ORIGEN-ARP) and Monte Carlo N-Particle (MCNP) codes were used to evaluate neutron flux on the suface of a PWR dry storage cask. ORIGEN-ARP was developed to satisfy a need for an easy-to-use standardized method of isotope depletion/decay analysis for spent fuel, fissile material, and radioactive material. It can be used to evaluate for spent fuel characterization, isotopic inventory, radiation source terms, and decay heat [3, 4]. MCNP code is a tool for particle transport calculations and widely used for evaluation of radiation shielding. It can be used for transport of neutrons, photons and electrons [5].

To define source term, ORIGEN-APR code was used to calculate neutron emission rate and energy spectra from a WH 17 x 17 type spent fuel. The neutron emission rate varied depending on radiological characteristics which are enrichment, burnup rate, and cooling time. The ranges of enrichment and burnup rate of spent fuel in Korea were 2-5 wt% and 25,000-50,000 MWD/MTU, respectively [6]. In this study, the calculation was performed using above ranges.

Axial burnup distribution also should be considered to evaluate neutron flux of a dry storage cask. Axial burnup distribution will be flatted over time due to the fuel depletion and the buildup of fission product that occurs near the center of fuel [7]. Generally, The gamma-ray emission rate is linearly proportional to the axial burnup distribution. The neutron axial burnup distribution is proportional to the 4.0-4.2 square of the gamma-ray axial burnup distribution [8]. Figure 1 shows axial burnup distribution of neutron and gamma source.

In this study, TN-32 dry storage cask was selected to evaluate neutron flux on the surface of a cask. TN-32 dry storage cask is a canister-based cask which was developed by AREVA-TN [9]. TN-32 dry storage cask consists of carbon steel overpack, neutron shield, trunnion, and etc. Geometry and material properties of the spent fuel assemblies and casks were simulated using MCNP code. It is assumed that thirty-two PWR spent fuel assemblies were loaded in a cask. Spent fuel assemblies were simulated assuming that were homogenized.

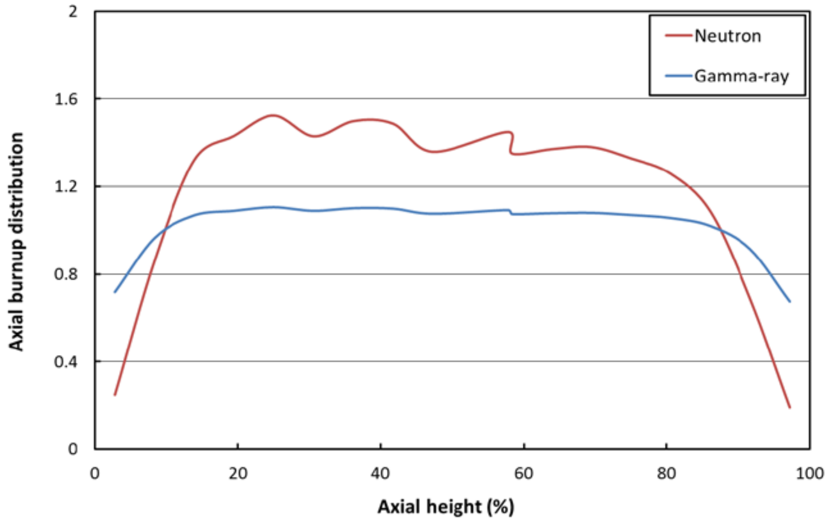

Figure 1. Axial burnup distribution of neutron and gamma source

Figure 2 shows the evaluation points of neutron flux in this study. It was divided into top, side, and bottom. The height of a TN-32 dry storage cask was about 460 cm, and there were 27 points at intervals of 20 cm to evaluate neutron flux on the surface of a cask. In case of the upper and lower of a TN-32 dry storage cask, the distance from the center of the cask to the side surface was 90 cm. There were 6 points at intervals of 20 cm to evaluate neutron flux on the surface of a cask.

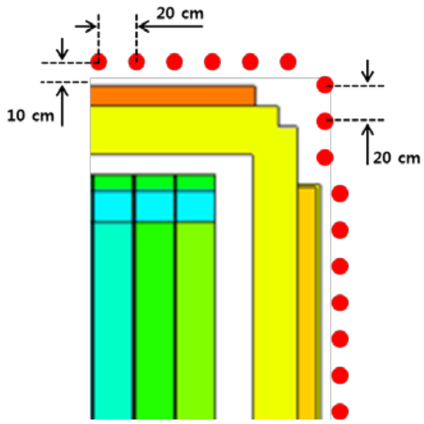

Figure 2. The evaluation points of neutron flux in a cask

RESULTS AND DISCUSSIONS

Figure 3 shows the neutron emission rate and energy spectra of PWR spent fuels in a dry storage cask. The neutron emission rate and energy spectra were calculated considering radiological characteristics of spent fuels generated in Korea. For conservative evaluation the enrichment, burnup rate, and cooling time were 4.5 wt%, 45,000 MWD/MTU, and 10 years, respectively. The neutron emission rate gradually increased along with neutron energy up to 3 MeV, and then decreased.

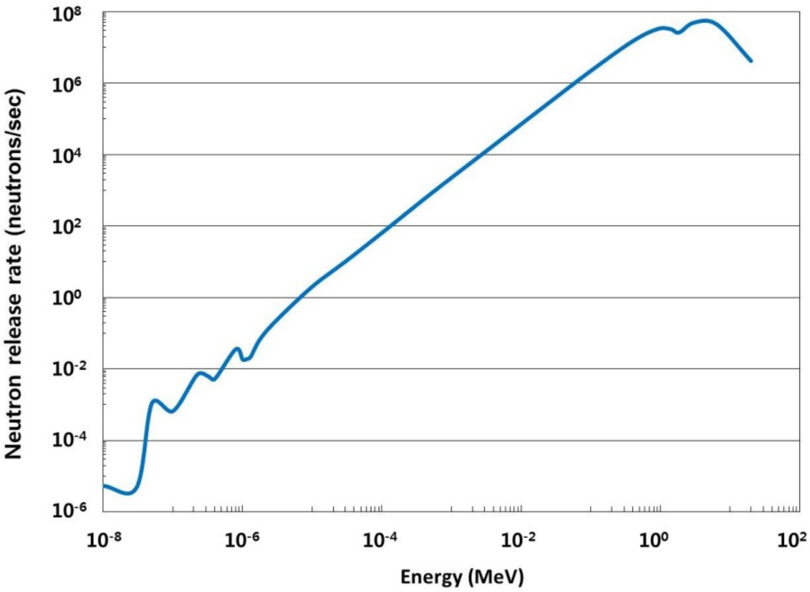

Figure 3. Neutron emission rate and energy spectra of PWR spent fuels in a dry storage cask.

MCNP modeling of TN-32 dry storage cask is given in figure 4. TN-32 dry storage cask was simulated considering carbon steel overpack, neutron shield, trunnion, and etc. Thirty-two PWR spent fuel assemblies were also simulated by MCNP code.

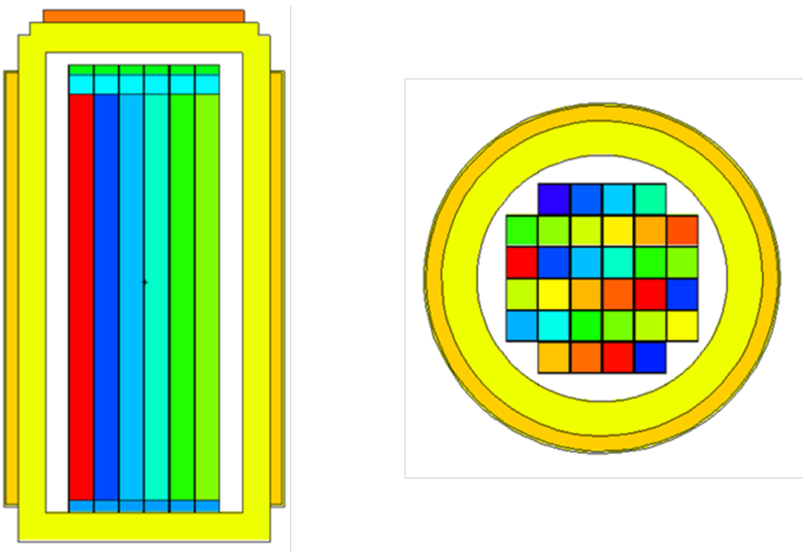

Figure 4. MCNP modeling of TN-32 dry storage cask

Neutron fluxes for TN-32 were calculated on the surface of top, side and bottom of a cask. Table 1 shows the average neutron fluxes which were calculated in this study. The average neutron fluxes on the surface of top and bottom were higher than on the surface of side.

Table 1. Average neutron flux on the surface of a dry storage cask

The evaluation point of neutron flux by cask height	The average neutron flux on the surface (neutrons/cm^2·sec)
Top	1.08 x 10^3
Side	4.50 x 10^2
Bottom	1.04 x 10^3

Figure 6 shows neutron flux on the surface of TN-32 dry storage cask by cask height. Neutron fluxes on the surface of side of a cask were generally similar on the midium height. However, neutron fluxes on the surface of top and bottom of a cask were relatively higher than those on the surface of side of a cask.

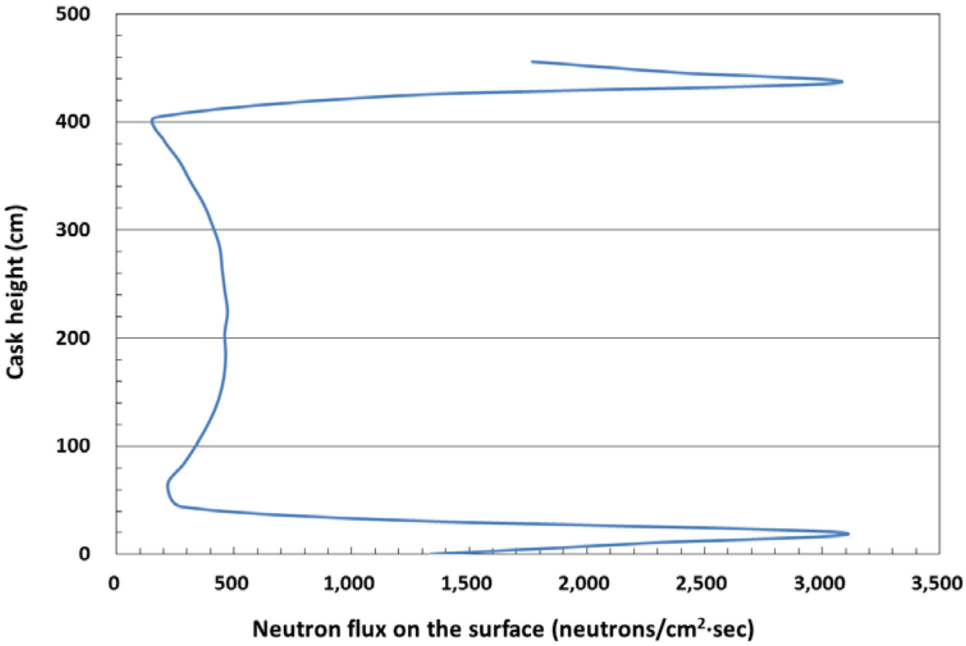

Figure 6. Neutron flux on the surface of TN-32 dry storage cask by cask height

Axial neutron flux distributions on the side part varied with cask structure. The reason for high neutron fluxes on the surface of top and bottom is that neutron shields were designed to have little effect on the top and bottom. Figure 7 shows the evaluation points indicating high neutron flux.

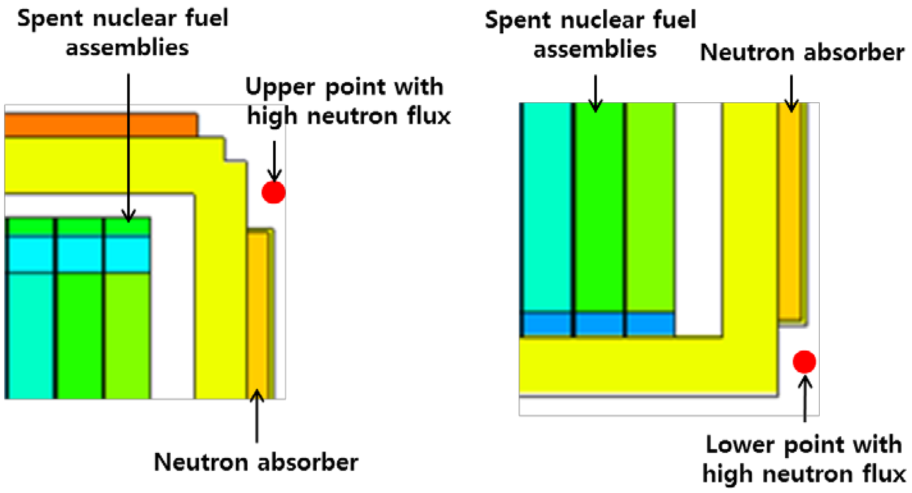

Figure 7. The evaluation points indicating high neutron flux

Figure 8 shows neutron flux on the surface of top and bottom of TN-32 dry storage cask by distance from center of a cask. Neutron fluxes on the surface of bottom were higher than on the surface of top. Neutron fluxes were gradually decreased as far from the center of a cask.

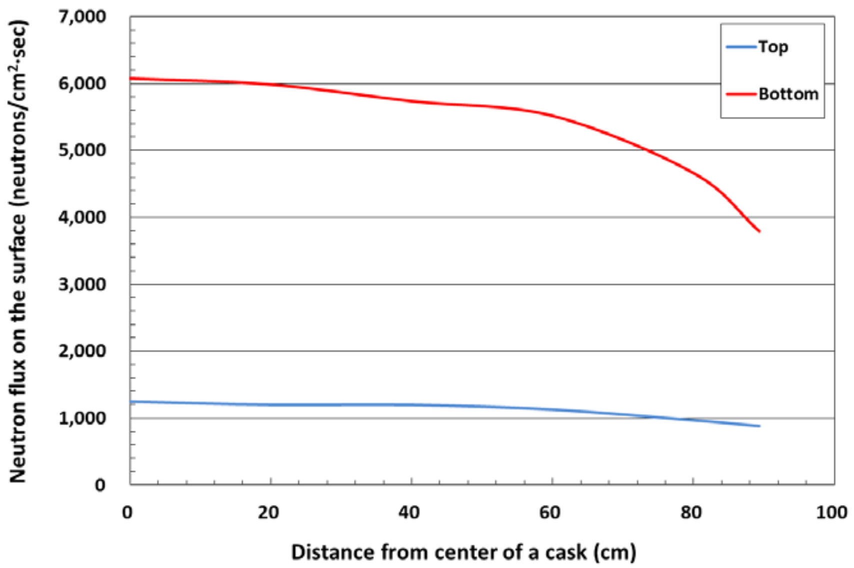

Figure 8. Neutron flux on the surface of top and bottom of a cask by distance from center of a cask

CONCLUSIONS

In this study, we developed an evaluation technique based on the neutron detection to verify the safety of spent fuels in a dry storage facility. The objective of this study was to evaluate neutron flux on the surface of a PWR dry storage cask. ORIGEN-ARP and MCNP codes were used to evaluate neutron flux on the suface of a PWR dry storage cask.

Neutron fluxes for TN-32 were calculated on the surface of top, side and bottom of a cask after 10 years of cooling time. For TN-32 dry storage cask, average neutron fluxes were evaluated to be 1.08 x 10^3 neutrons/cm^2·sec, 4.50 x 10^2 neutrons/cm^2·sec, and 1.04 x 10^3 neutrons/cm^2·sec on the surface of top, side, and bottom of a cask, respectively. Axial neutron flux distributions on the side part varied with cask structure. The reason for high neutron fluxes on the surface of top and bottom is that neutron shields were designed to have little effect on the top and bottom.

This study can be applied to evaluate the characteristics of spent fuels and neutron flux of a dry storage cask. Furthermore, this study can be conducted with measurement data from neutron flux of a dry storage cask.

ACKNOWLEDGEMENTS

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KoFONS), granted financial resource from the Nuclear Safety and Security Commission (NSSC), Republic of Korea (No. 1503006).

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