Evaluation of Neutron Flux on the Surface of a PWR Dry Storage Cask – 17574

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ABSTRACT

Since the construction of a dry storage facility for spent nuclear fuel is expected in Korea, several methods have been studied for assuring the safety of pressurized-water reactor (PWR) spent fuels in dry storage casks. Existing evaluation methods are mostly based on detection technology of gamma-rays. However, the gamma-ray detection methods have high uncertainties due to background radiation. Therefore, we developed an evaluation method for the spent nuclear fuel monitoring based on neutron detection [1].

In this study, we evaluated neutron flux on the surface of a PWR dry storage cask using computational simulation. Neutron release rate and energy spectra from a Westinghouse (WH) 17x17 type spent fuel with 10 years cooling time were calculated using ORIGEN-ARP code. MCNP code was used to simulate geometry and material properties of the spent fuel assemblies and casks. Neutron flux after 10 years of cooling time of the spent fuels was calculated on the surface of top, side, and bottom of the casks. The metal cask and the concrete cask, called as KORAD-21 casks which newly developed by Korea Radioactive Waste Agency (KORAD), were selected for this study.

As a result, for the metal cask, average neutron flux were 4.79×10^2 neutrons/cm^2 sec, 5.29×10^2 neutrons/cm^2 sec, and 1.47×10^3 neutrons/cm^2 sec on the surface of top, side, and bottom of the cask, respectively. For the concrete cask, average neutron flux on the surface of top, side and bottom of the casks were 5.07×10^1 neutrons/cm^2 sec, 1.18×10^1 neutrons/cm^2 sec, and 1.46×10^3 neutrons/cm^2 sec, respectively. The study results can be applied to evaluate characteristics of PWR spent fuels and the neutron flux of dry storage cask.

INTRODUCTION

There are twenty-three operation nuclear power plants in Korea currently. PWR spent nuclear fuels generated were being stored in the temporary storage facilities in the nuclear power plants. The present, 760 metric tons uranium (MTU) of spent nuclear fuel are added to this inventory each year. Therefore, the storage capacity of the spent nuclear fuels in the temporary storage facilities will be saturated by 2024 [2]. In order to solve the saturation problem of the spent fuel storage, the interim storage facility to store spent nuclear fuel safely needed. The dry storage system has been proven in terms of safety technology and was internationally commercialized. Considering the domestic situation, dry storage system is planned to be constructed by 2035 and operated.

The several methods have been studied for assuring the safety of PWR spent fuels in dry storage casks, in Korea. Existing evaluation methods are mostly based on detection technology of gamma-rays. However, the gamma-ray detection methods have high uncertainties due to background radiation. Therefore, we developed the safety evaluation method by computational analysis based on neutron detection.

MATERIALS AND METHODS

ORNL isotope generation and depletion-Automatic Rapid Processing (ORIGEN-ARP) and Monte Carlo N-Particle (MCNP) codes were employed for this computational analysis. ORIGEN-ARP code is a calculation code for analyze spent fuel characterization and source term generation [3]. MCNP code is a general Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled particles transport [4]. Therefore, MCNP code is widely used for evaluation of radiation shielding and flux.

To evaluate neutron flux on the surface of the dry storage cask, the neutron release rate of spent nuclear fuel needs. The neutron release rate varied depending on radiological characteristics of spent nuclear fuel. Radiological characteristics consist of enrichment, burnup rate, and cooling time. Enrichment and burnup rate range of the nuclear fuel used in Korea was 2-5 wt%, and 25,000-50,000 MWD/MTU, respectively. Therefore, neutron release rate and energy spectra of spent nuclear fuel in consideration of radiological characteristics were calculated using ORIGEN-ARP code.

To evaluate neutron flux of dry storage cask, the axial burnup distribution should be also considered. Fig. 1 shows axial burnup distribution of neutron and gamma source. Axial burnup distribution will be flatted over time due to the fuel depletion and fission product buildup that occurs near the center of fuel, because it will deplete fuel near the axial center at a greater burnup rate than at the edge [5]. The gamma-ray release rate is generally directly 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 [6].

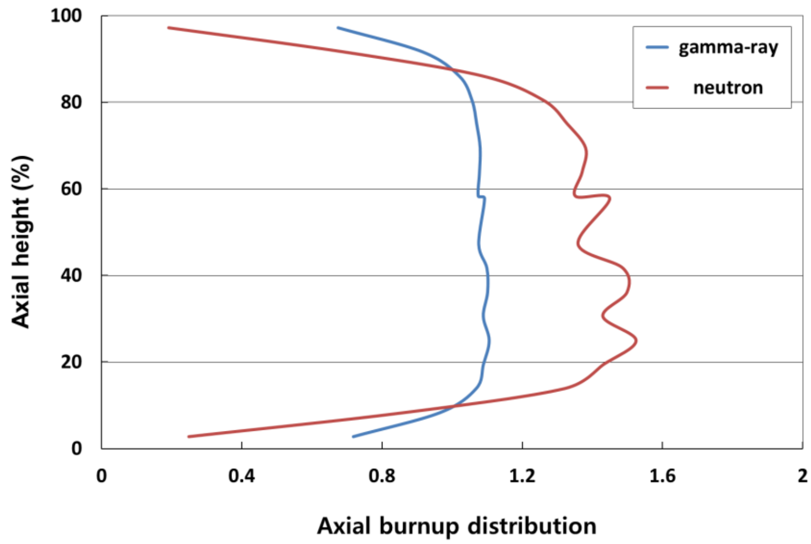

Fig. 1. Axial burnup distribution of neutron and gamma source

The dry storage casks can be classified as a metal cask and concrete cask, called as KORAD-21 casks. The metal cask was developed for both transport and storage purposes, and the concrete cask was developed only for storage purpose. The metal cask consists of canister containing the spent nuclear fuel assemblies, the neutron absorber, the heat transfer pins, trunnion, and etc. The concrete cask consists of the neutron absorber, air ducts, top and bottom cover, canister containing the spent nuclear fuel assemblies, and etc. Therefore, geometry and material properties of the spent nuclear fuel assemblies and casks were simulated using MCNP code. It is assumed that the same twenty-one PWR spent fuel assemblies were loaded. Then, spent nuclear fuel assemblies were simulated assuming that were homogenized. Complex geometries, such as top and bottom structural material of fuel assembly, were simply simulated.

The flux evaluation point was selected for evaluation of neutron flux of dry storage casks (see Fig. 2). It was divided into top, side, and bottom. The height of a metal cask was 530 cm, and the evaluation points were selected for 28 points at intervals of 20 cm. In case of the upper and lower of a metal cask, the distance from the center of the casks to the side surface was 110 cm. So, the evaluation points were selected for 6 points at intervals of 20 cm, respectively. The height of a concrete cask was 580 cm, and the evaluation points were selected for 30 points at intervals of 20 cm. In case of the upper and lower of a concrete cask, the distance from the center of the casks to the side surface was 160 cm. So, the evaluation points were also selected for 9 points at intervals of 20 cm, respectively.

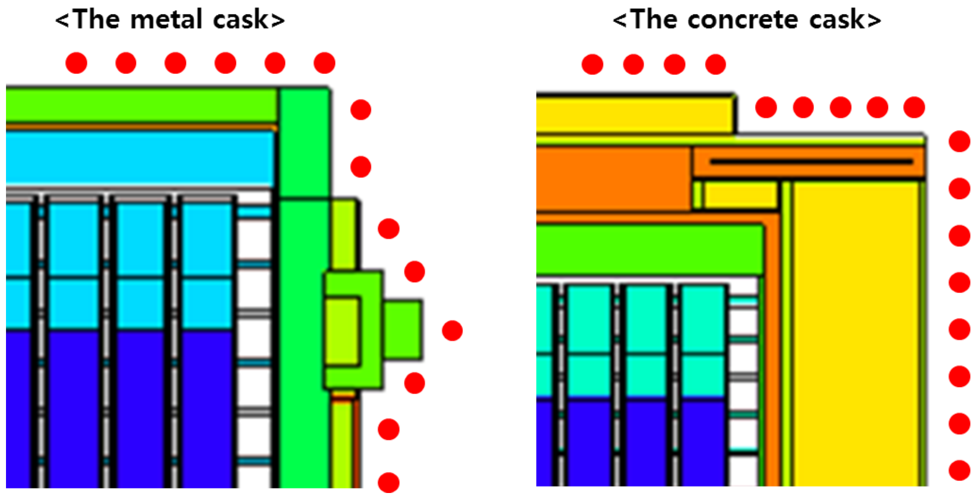

Fig. 2. The flux evaluation point of the dry storage cask

RESULTS AND DISCUSSIONS

Fig. 3 shows the neutron release rate and energy spectra of PWR spent nuclear fuels in dry storage cask. Considering the radiological characteristic of Korea nuclear fuels, the neutron release rate and energy spectra were calculated. The enrichment, burnup rate, cooling time, and assembly type for conservative evaluation were 4.5 wt%, 45,000 MWD/MTU, 10 years, and WH 17x17 assemblies, respectively. The neutron release rate gradually increased along with neutron energy up to 3 MeV, and the decreased.

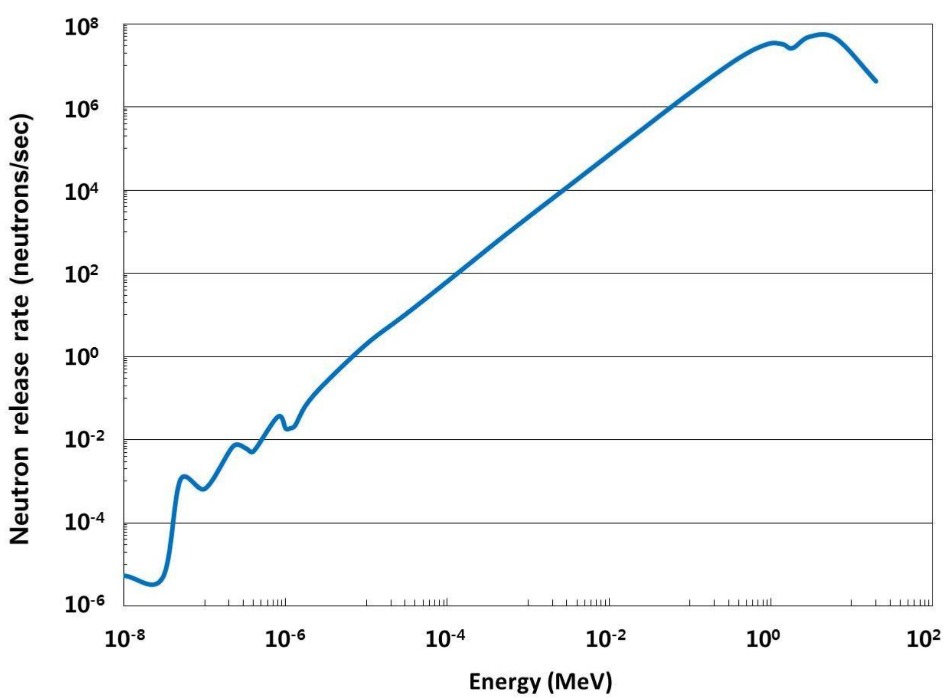

Fig. 3. Neutron release rate and energy spectra of PWR spent nuclear fuels

The canister contained the 21 spent nuclear fuel assemblies and the neutron shielding was simulated in common. The metal cask was simulated by considering heat transfer pins, trunnion, and the concrete cask was also simulated to air ducts, top and bottom cover. MCNP modeling of KORAD-21 dry storage cask is given in Fig. 4, and Fig. 5.

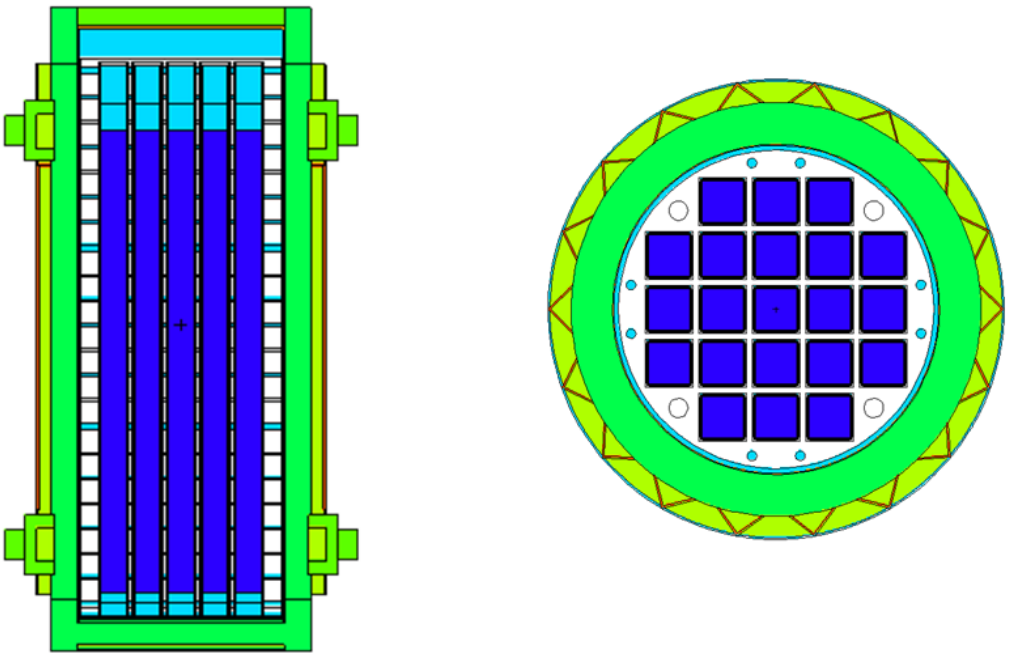

Fig. 4. MCNP modeling of KORAD-21 metal dry storage cask

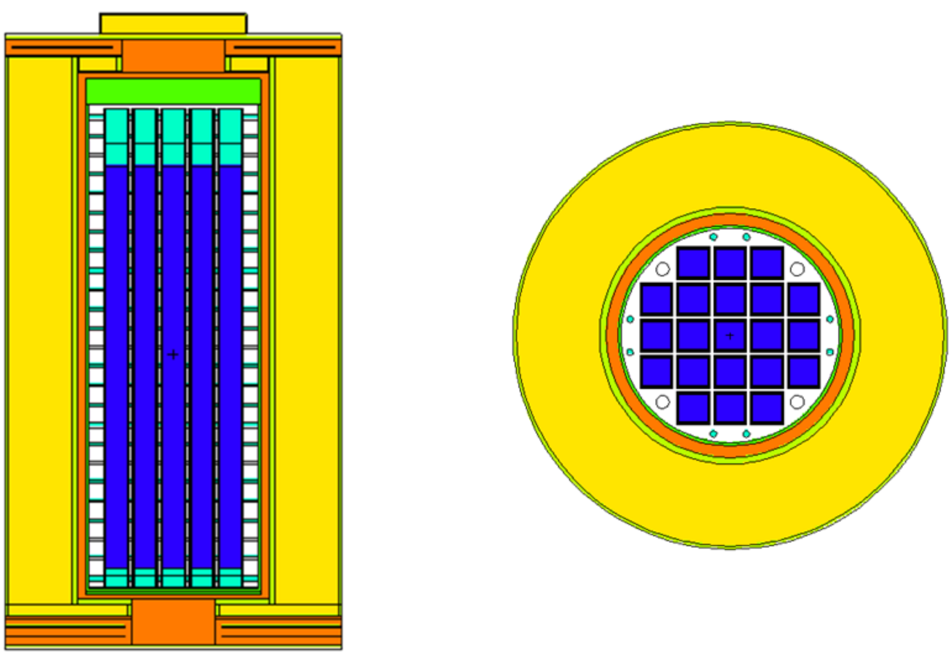

Fig. 5. MCNP modeling of KORAD-21 concrete dry storage cask

TABLE I shows the average neutron flux on the dry storage cask surface of top, side, and bottom. The average neutron flux on the side surface of the metal cask was higher than the concrete cask. This is because neutron shield of the concrete cask was thicker than the metal cask. Also, average neutron flux on the surface of bottom was higher than side and top. This is because the distance from the spent nuclear fuel to the top surface is longer than the distance to the bottom surface.

TABLE I. The average neutron flux on the dry storage cask

The flux evaluation point	**The average neutron flux on the surface (neutrons/cm^2 sec)**	
	The metal cask	**The concrete cask**
Top	4.79×10^2	5.07×10^1
Side	5.29×10^2	1.18×10^1
bottom	1.47×10^3	1.46×10^3

The neutron flux on the surface of dry storage cask is represented Fig. 6, and Fig. 7. For the metal cask, the neutron flux distribution by axial direction was the lowest at the surface on trunnion, which worked as more shielding. And, the highest neutron flux at lower part was because of less shielding. For the concrete cask, the axial neutron distribution showed especially higher values at around top and bottom parts. They could be attributed to air ducts for natural convection, which resulted from less shielding.

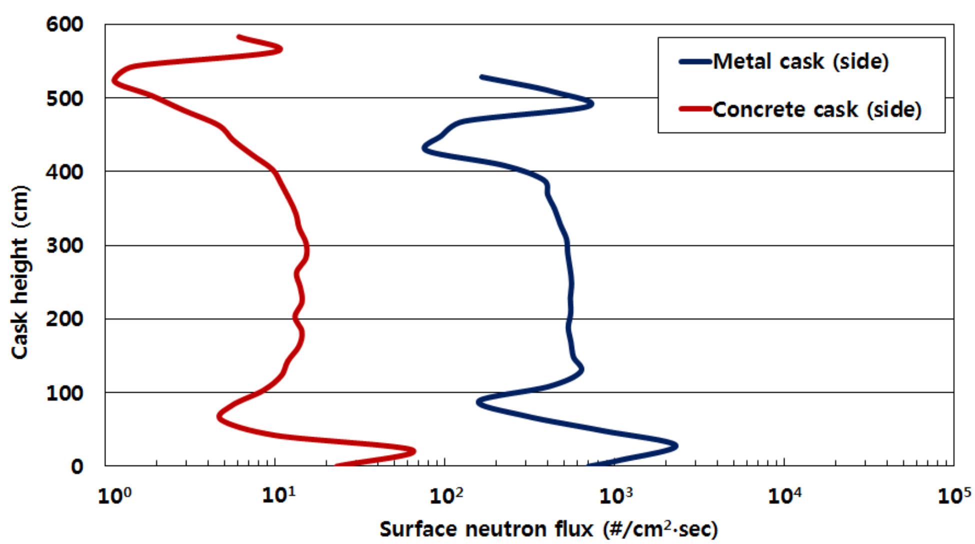

Fig. 6. Neutron flux on the surface of side of dry storage cask by cask height

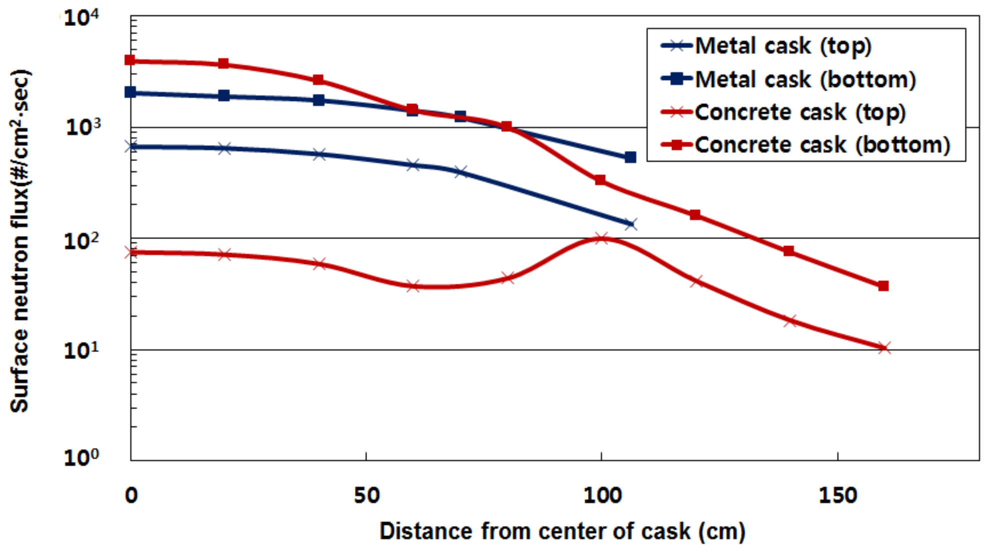

Fig. 7. Neutron flux on the surface of top and bottom of dry storage cask by distance from center of cask

CONCLUSIONS

Continuously developing a new effective method is essential to secure spent nuclear fuel in a difficult-to-access area. Therefore, we proposed an evaluation method for the spent nuclear fuel monitoring based on neutron detection. Based on the proposed method, we evaluated the average neutron flux on the surface of dry storage casks using computer code. The neutron flux varies depending on the cask material and the structure of dry storage cask. The neutron flux on the side surface of the metal cask was higher than the concrete cask. Also, average neutron flux on the surface of bottom was higher than side and top. For the metal cask, average neutron flux were 4.79×10^2 neutrons/cm^2 sec, 5.29×10^2 neutrons/cm^2 sec, and 1.47×10^3 neutrons/cm^2 sec on the surface of top, side and bottom of the casks, respectively. For the concrete casks, average neutron flux were 5.07×10^1 neutrons/cm^2 sec, 1.18×10^1 neutrons/cm^2 sec, and 1.46×10^3 neutrons/cm^2 sec on the surface of top, side, and bottom of the casks, respectively.

This study can be applied to evaluate characteristics of PWR spent nuclear fuels and the neutron flux on the surface of the dry storage cask. The results of this study will be used for comparative analysis of neutron measurement data from PWR spent fuels in dry storage cask.

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