# Estimation of Neutron Detection Rates from Spent Nuclear Fuel

Kwangheon Park[1)], Jinhyun Sung[1)], Hyemin Park[2)], Moonoh Kim[3)], Yangsoo Song[2)]
*1)Kyung Hee University, 2)ORION EnC, 3)KONES Corporation*
*E-mail: kpark@khu.ac.kr*

***Keywords : Neutron detection, $^3$He detector, MCNP, ORIGEN-ARP**

## 1. Introduction

As wet storage facilities at nuclear power plants approach their maximum capacity, transferring spent nuclear fuel (SNF) to dry storage has become an inevitable requirement, prompting extensive preparatory efforts [1]. Ensuring a safe and successful transfer necessitates a verification process to confirm that the physical characteristics of the transferred SNF match its documented records. Furthermore, this type of experimental verification plays a crucial role as a nuclear safeguards tool in the context of non-proliferation [1]. To meet this need, ORION-EnC, KONES Co., and Kyung Hee University have jointly developed a new SNF verification system. Prior to its scheduled empirical validation using actual SNF at the Korea Atomic Energy Research Institute (KAERI), this paper computationally predicts the expected $^3$He neutron detection rates for the targeted fuel.

## 2. The process for the estimation

First, we introduce the newly developed verification system. Next, we describe the computational modeling of the neutron emission from the SNF and the subsequent $^3$He detection process using Origen-ARP and MCNP. We then present a preliminary validation experiment using a $^{252}$Cf neutron source, comparing MCNP predictions with actual measurement results. Finally, based on this validated modeling approach, we predict the expected neutron detection rates for the targeted SNF to be measured at KAERI.

### 2.1. The burnup verification system developed

The developed burnup verification system centers on a cylindrical $^3$He detector (LND 3He-2528). To maximize detection efficiency by moderating incident fast neutrons, the detector is surrounded by a high-density polyethylene (HDPE) moderator and combined with a CZT detector for gamma-ray measurement. The entire module is sealed within an aluminum housing to ensure watertightness and structural integrity in harsh wet storage environments. Figure 1 shows the system's geometric design optimized for simultaneous neutron and gamma detection. Fig. 1(a) presents a photograph of the assembled system, and Fig. 1(b) illustrates the cross-sectional arrangement of the central $^3$He detector within the HDPE and a peripheral CZT detector.

### 2.2. Model for detection rate estimation

In spent nuclear fuel (SNF), neutrons generated via spontaneous fission and (α,n) reactions undergo subcritical multiplication by actinides or absorption by fission products. Emitted neutrons from SNF pass through water shielding and are measured by a $^3$He detector, as illustrated in Figure 2.

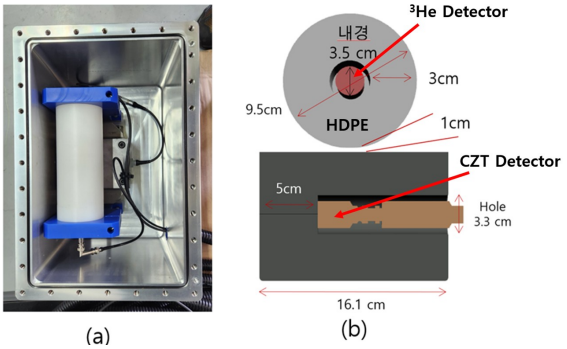



**Figure 1. (a) the assembled system, (b) the cross-sectional arrangement of the central $^3$He detector within the HDPE and a peripheral CZT detector.**

Using MCNP, we calculate the detection probability ($\mathbf{P_{Det}}$) of a single neutron originating in the SNF and undergoing an (n,p) reaction within the detector. To accurately model neutron interactions within the fuel, ORIGEN-ARP is utilized to calculate the inventories of 12 key actinides and 16 important fission products, which then serve as material definitions in the MCNP simulations.

Additionally, ORIGEN-ARP computes the SNF neutron source intensity to determine the neutron emission rate per fuel assembly ($\mathbf{S_{FA}}$). Ultimately, the expected neutron detection rate of the $^3$He detector is obtained by multiplying $\mathbf{P_{Det}}$ and $\mathbf{S_{FA}}$.

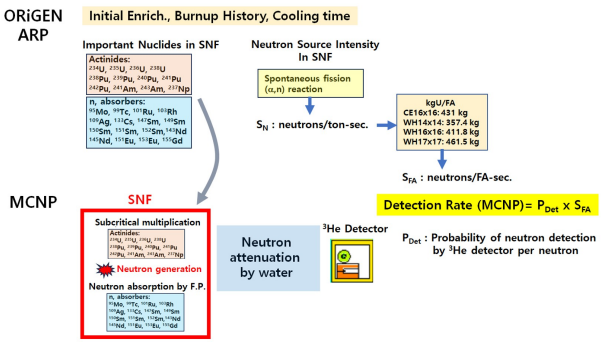



**Figure 2. Calculation model for detection rate using MCNP and ORIGEN-ARP.**

### 2.3. Validation of the MCNP model using $^{252}$Cf

To validate the computational model, neutron measurement experiments were conducted using a $^{252}$Cf

source positioned at various distances from the submerged detector within a 1-m-diameter metal tank filled with over 80 cm of water. This setup was simulated using MCNP to derive the expected detection rates. Figure 3 illustrates the experimental apparatus, the measured signals, and a comparison between the experimental and predicted values. The detected peak precisely corresponds to the 764 keV (n,p) reaction energy. Both the total and peak detection rates were evaluated against the MCNP predictions. The measured total detection rate averaged 90% of the MCNP estimate, and this ratio was adopted as the baseline for predicting the actual expected detection rates.

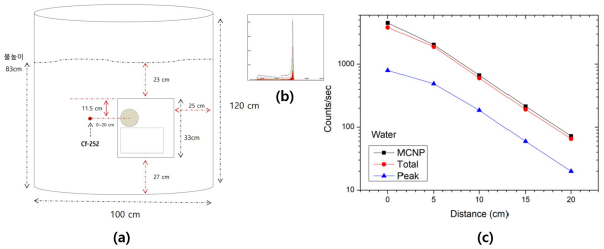



**Figure 3. (a) validation experiment setup (b) measured detector signal (c) comparison of measured and calculated detection rates.**

### 2.4. Estimation of detection rate from SNF

Four different SNF assemblies (J44, C15, A39, and A17) were selected for the measurement. The initial enrichment, discharge burnup, cooling time, and operating history of each assembly were comprehensively reviewed [2]. Based on these parameters, the inventories of 28 key nuclides within the SNF were calculated using ORIGEN-ARP and subsequently incorporated into the MCNP model as material definitions. Additionally, the total neutron emission rate per fuel assembly ($S_{FA}$) was derived from the ORIGEN-ARP neutron source intensity calculations.

Figure 4 illustrates the MCNP geometry model developed to evaluate the detection probability ($P_{Det}$). As shown in the model, four detection devices were arranged axially, with a 50 cm vertical spacing between each detector relative to the center. To assess the spatial dependence of the detection rate, four different distances (1.7, 6.7, 11.7, and 21.7 cm) between the SNF and the detectors were simulated. Furthermore, the axial burnup profile within the MCNP model was precisely defined based on actual axial gamma scanning data obtained from the J44 assembly.

Figure 5 illustrates the estimated neutron detection rates for the four SNF assemblies. Axially, the rate remains relatively constant up to 100 cm from the center, with a slight decrease at +150 cm reflecting the drop in the axial burnup profile. Radially, the detection rate exhibits exponential attenuation and is highly sensitive to the stand-off distance; a mere 1 cm variation causes an approximately 20% change. Additionally, absolute detection counts vary significantly among the assemblies -with J44 showing the highest rates and A17 the lowest- due to their distinct burnup histories and cooling times. Consequently, precisely maintaining the radial distance between the SNF and the detector is critical during actual field measurements.

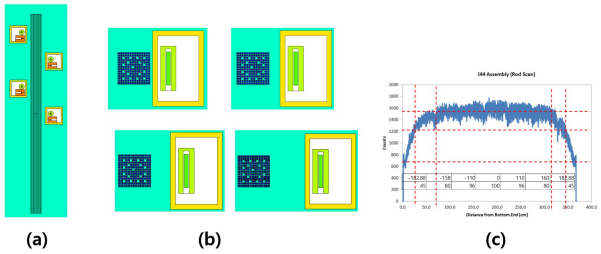



**Figure 4. MCNP model for detection rate.**

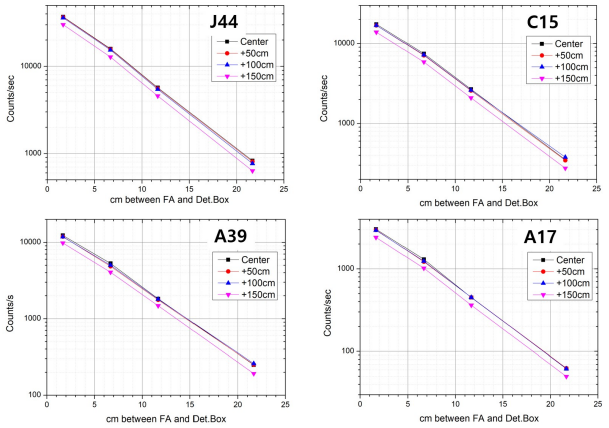



**Figure 5. Estimated detection rate of SNF assemblies.**

## 3. Conclusion

A newly developed SNF verification system is scheduled for empirical validation at KAERI. To prepare for these field measurements, a computational framework using ORIGEN-ARP and MCNP was established and validated using $^{252}$Cf experimental data. This validated model was used to predict the expected $^{3}$He neutron detection rates for the targeted SNF assemblies in KAERI, providing a crucial baseline for the upcoming application of the developed verification system.


## Acknowledgement

This work was supported by the Korea Institute of Energy Technology and the Ministry of Climate, Energy and Environment of the Republic of Korea (RS-2022-KP002854)


## REFERENCES


[1] Review of Information for spent nuclear fuel burnup confirmation, NUREG/CR-6998

[2] 원자력연구원, 사용후 핵연료 효과적용을 위한 연소도 분포 측정기술개발 최종보고서 (2012.6)