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A Study on Enhancing Burnup Estimation Accuracy for Spent Nuclear Fuel Assemblies with Radial Burnup Distribution

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1. Introduction

Our research group is developing a detection system designed to estimate the burnup of spent nuclear fuel assemblies by utilizing gamma-ray spectra collected from multiple detectors placed at various locations. We have enhanced the system's design to allow for simultaneous acquisition of gamma-ray spectra from these multiple detectors, and subsequently compared its performance in estimating average burnup with that of conventional systems. Conventional methods typically rely on gamma-ray count rates measured at a single location or isotopic ratios of specific radionuclides, which can introduce inaccuracies. Although the radial burnup distribution in spent fuel assemblies tends to be more uniform than the axial distribution, it can become non-uniform due to design parameters and fuel loading scenarios during reactor operation. While the axial burnup distribution can be measured by adjusting the detector position, radial burnup distribution is more challenging to measure, potentially leading to errors in estimating the average burnup. Preliminary findings suggest that our system provides more accurate burnup estimations, particularly in cases where radial burnup asymmetry is more pronounced.

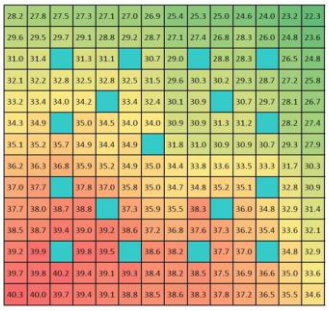

Fig. 1. Radial burnup profile of 14x14 spent fuel assemblies based on operator declared information[1]

2. Methods and Results

Figure 2 presents the overall flowchart of this study, showing the process of comparing the estimated average burnup calculated by combining source term evaluation of spent nuclear fuel assemblies with radiation detector response simulations to the actual average burnup used as input.

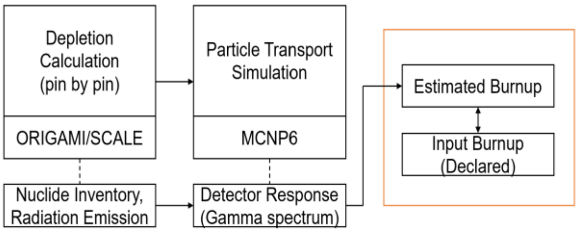

Fig. 2. Flowchart of the research

2.1 Source term evaluation using ORIGAMI

ORIGAMI is a burnup calculation module integrated within the reactor analysis tool SCALE, and it can be used to evaluate changes in nuclide inventory during reactor operation and the resulting radiation emission characteristics. Unlike conventional burnup calculation modules, ORIGAMI allows for different burnup calculations for individual fuel rods, which means it can account for the effects of burnup variations based on the reactor's power distribution [2].

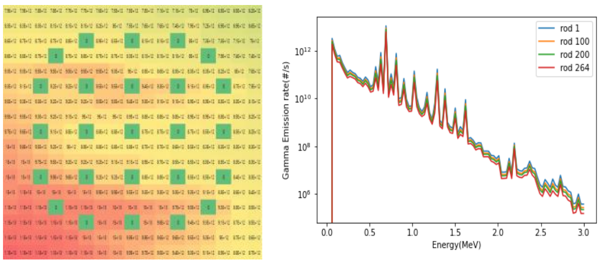

Fig. 3. Total gamma intensity distribution and gamma energy emission spectrum by individual fuel rods

For the source term evaluation of spent nuclear fuel assemblies, the nuclide inventory and emitted gamma-ray spectrum were derived based on key variables of the spent fuel assembly, such as initial enrichment, burnup, and cooling time. The calculated results are used as source term inputs for the detector response simulation of the detection system. Specifically, as shown in Figure 1, burnup calculations were performed by applying different burnup conditions to individual fuel rods in order to account for non-uniform radial burnup distribution. Through this process, the total gamma intensity and gamma-ray emission spectra for each fuel rod were derived, as illustrated in Figure 3.

2.2 Burnup estimation by detector response simulations

The detection system was modeled using MCNP, a widely recognized particle transport simulation code. In the modeling process, the positioning of each detector and the geometric configuration of the spent nuclear fuel assembly were accounted for to accurately simulate particle transport. Using the developed detection system model, MCNP simulations were conducted to replicate the response of each detector. The gamma-ray energy spectra from each detector were obtained by incorporating the previously derived source term evaluation of the spent nuclear fuel assembly.

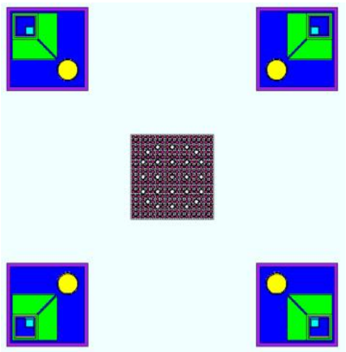

Fig. 4. Simulation model of detection system and spent fuel assembly.

These gamma-ray energy spectra were subsequently analyzed to estimate the average burnup of the spent nuclear fuel assembly. The estimated burnup using a single detector was compared with that obtained from multiple detectors, considering variations in the radial burnup distribution of the spent fuel assembly. When the radial burnup distribution is as shown in Figure 3, the average burnup estimated by the measurement system showed an error of approximately 36% compared to the operator-reported average burnup. In contrast, under the same conditions, the average burnup estimated using all four detectors was 31 GWd/MTU, with an error of about 3.6%. This indicates that using the system proposed by our research group can improve the accuracy of burnup estimation compared to using the conventional measurement system for estimating the average burnup of spent nuclear fuel assemblies.

3. Conclusions

It has been demonstrated that utilizing gamma-ray spectra from multiple detectors provides a more accurate estimation of the average burnup in spent nuclear fuel assemblies compared to relying on a single detector. While a single detector may be adequate when the radial burnup distribution is uniform, accurate estimation becomes challenging when the gamma-ray emission distribution aligns with the radial burnup distribution, as illustrated in Figure 3. The accuracy of burnup estimation can be significantly enhanced by incorporating signals from multiple symmetrically arranged detectors.

ACKNOWLEDGEMENTS

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