CO3-1

Reactor noise measurement using ex-core detector system

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INTRODUCTION:

Reactor noise measurements targeting KUR were conducted from FY2014 to FY2021 using fission chambers for startup monitoring system as in-core detectors and He-3 detectors as ex-core detectors. However, because thermal neutrons were the primary measurement target, sufficient results could not be obtained due to various challenges, including the distance between the reactor core and the detectors [1, 2]. In this study, epithermal neutrons were selected as the measurement target, and the analysis methodology was also improved. Reactor noise measurements of KUR were therefore carried out again. The detectors used were BF_3 neutron detectors equipped with Cd covers, and measurements were performed at the CN-1 and B-3 experimental holes. The results of this study contribute to the advancement of nuclear characteristics analysis for high-power research reactors.

EXPERIMENT:

The detector system used in this study, referred to as "ERYNGII" [3], is designed to efficiently measure epithermal neutrons. In this system, thermal neutrons are shielded by a cadmium (Cd) plate, while epithermal neutrons are moderated by polyethylene, enabling effective detection of epithermal neutron flux. This detector system was installed at the front of CN-1 and B-3 experimental holes, and time-series data of the epithermal neutron flux were acquired. The measurements were conducted during nighttime on Wednesdays, when the concentration of Xe-135 in the fuel had reached saturation and control rod operations due to automatic operation of KUR were minimal.

RESULTS:

Figure 1 shows measured time-sequence data. Analysis of the time-sequence data were carried out using the variance-to-mean method in an attempt to derive the prompt neutron decay constant (β_{eff}/l). As shown in Table 1, it is found that the variance-to-mean analysis could not be successfully performed for any of the measurement datasets obtained from CN-1 and B-3. The calculated values were obtained using MVP3 [4] with JENDL-5 [5], and account for the average burnup. Further analyses using the Rossi-α method and the autocorrelation function method are planned for future work.

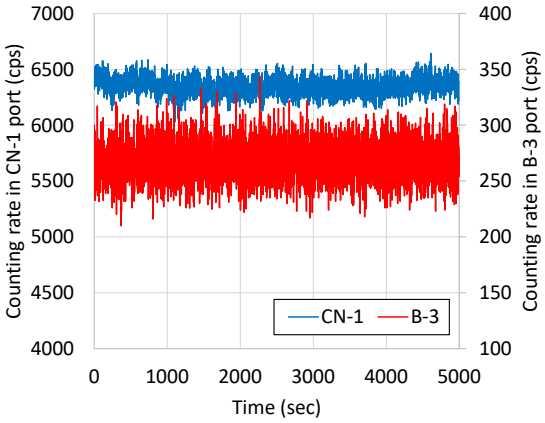

Figure 1 Measured time-sequence data

Table 1 Obtained prompt neutron decay constant

CN-1	B-3	Calc.
48.3 (/s)	Non	114.3 (/s)

[1] S. Hohara, *et al.,* KURNS ProgressReport2017, CO3-1, (2018).
[2] S. Hohara, *et al.,* KURNS ProgressReport2021, CO3-10, (2022).
[3] T. Sano, *et al.,* KURNS ProgressReport2021, CO3-6, (2022).
[4] Y. Nagaya, et al., JAEA-Data/Code 2016-018, (2017).
[5] O. Iwamoto, *et al., J. Nucl. Sci. Technol.,* 60, 1-60 (2023).

Characterization of Spent Nuclear Fuel Discharged from Kyoto University Reactor

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INTRODUCTION: Nuclear properties of a spent fuel are determined by its isotopic compositions. They are often evaluated based on the burn-up of the fuel. The passive γ ray measurement (PG) has been considered effective to confirm the burn-up, as well as the radiation dose and the radiation toxicity, etc. In this work, the PG was tested for a spent fuel element discharged from Kyoto university reactor (KUR).

EXPERIMENTS: Spent fuel element FL-14 discharged from KUR 4.8 years ago was stored vertically in a rack equipped on the bottom of the spent fuel pool. The burn-up of FL-14 is 24 GWd/t. The air environment above the pool is shielded from γ ray exposure from FL-14 by layer of light water with 5.2 m in thickness. For PG with a HPGe placed above the pool, a pipe was immersed in the water as shown in Fig. 1. The lower end of the pipe was plugged so that inside the pipe was filled with air. γ rays only pass through the pipe can be counted. The inner diameter of the pipe was 28 mm to keep the radiation dose at the top end of the pipe to be around 2 μSv/h.

RESULTS: In Fig. 2, measured spectra are shown. The room back spectra was measured with the HPGe placing 1 m apart from top of the pipe. No fission product (FP) γ ray was measured, proving the shielding capability of the light water layer. Above the pipe, γ rays from FPs, ^{137m}Ba, ^{134}Cs, ^{152,154}Eu, ^{106}Rh and ^{144}Pr were measured as well as those from ^{60}Co. The total counting rate of γ rays was 2.7 kcps. One of the difficulties to determine the burn-up of FL-14 by the activity ratio of ^{134}Cs/^{137}Cs is less peak to base ratio of 605 keV γ ray from ^{134}Cs. That might be due to detection of scattered components of 662 keV γ rays from ^{137m}Ba. In the next measurement, we will try to store fuel elements horizontally to reduce the component. Since the cooling time after the discharge was within a several years, we can easily measure γ rays from ^{106}Ru($T_{1/2}$=1.02 y)→^{106}Rh($T_{1/2}$=30 s) and ^{144}Ce($T_{1/2}$=285 d)→^{144}Pr($T_{1/2}$=7.2 m). ^{106}Rh is known for its radiation toxicity so reprocessing of a spent commercial fuel assembly is prohibited before cooling time of 15 y. Quantification of the toxicity by measuring the γ ray would be useful to allow the cooling time reasonably shorter.

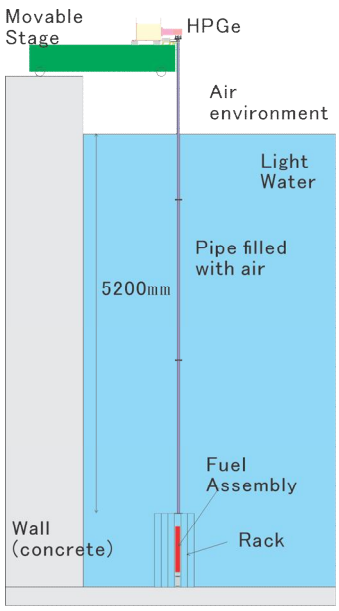

Fig. 1 Spent fuel pool and measurement geometry.

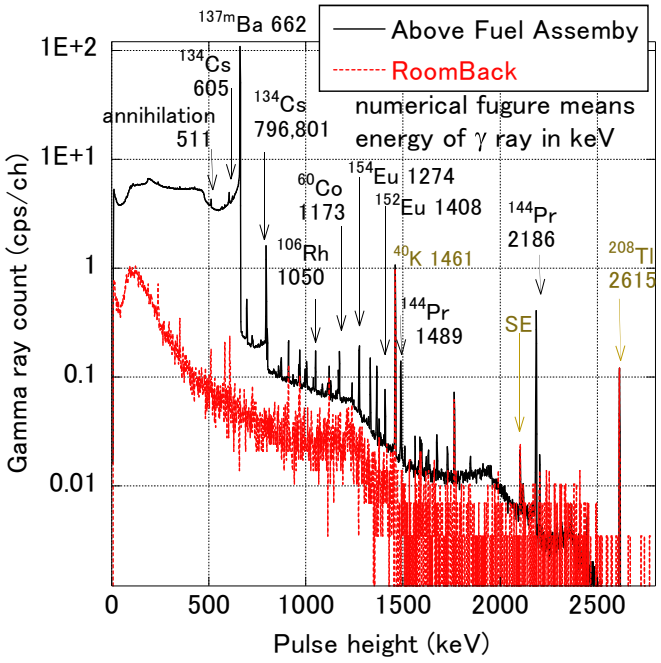

Fig. 2 Measured γ ray spectrum.

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Neutron flux measurement in sub-pile room for evaluating KUR facility activation

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INTRODUCTION: For the decommissioning of the KUR, it is important to evaluate the amount of radio activation materials such as the experimental equipment, the biological shield and so on. In general, the evaluation of the activation dose is mainly based on numerical calculations. In numerical calculations, the neutron spectrum and absolute neutron flux are important input information. However, the absolute values often differ by several 10% or more between the calculated and measured values. With that background, the research project has been conducting neutron flux measurements on the surface of biological shields and constructing a numerical calculational model [1]. Furthermore, neutron flux measurements required for radiation evaluations related to decommissioning have not been conducted at nearly all reactors, including research reactors. With this background in mind, this study has commenced neutron flux measurements on the surface of the KUR biological shielding and within the fuel storage facilities in the KUR reactor core tank using the foil activation method or neutron detectors, starting in fiscal year 2022. In the 2025FY, a neutron flux measurement was conducted in KUR sub-pile room.

EXPERIMENT: Neutron flux measurement in the sub-pile room were using by Au foils. One Au foil sample was 10 mm in diameter and 0.05mm thickness (approximately 75 mg). The foils were placed at 4 locations around the primary cooling water outlet piping in the sub-pile room shown in Fig.1. Neutron irradiation was conducted from Oct. 29 to 30 in 2025. KUR has been operated with 5MW in the period. After the irradiation, a gamma ray (411 keV) emitted from Au-198 were measured using a Ge detector.

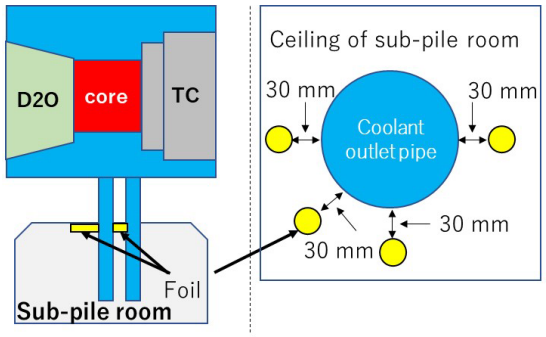

Fig.1 Location of Au foils setting

RESULTS: The averaged counting rate was 0.028 (cps) by the Ge detector. Where the all samples were measured simultaneously. The neutron flux was evaluated based on the measurement under the following assumptions:
1) The neutron flux was assumed to be thermal neutron flux, and self-shielding due to resonance absorption was neglected.
2) Attenuation of the neutron flux within the sample was neglected.
3) Perturbation of the neutron flux due to the placement of the sample was neglected.
4) The neutron temperature was assumed to be room temperature and the neutron capture cross section of Au-197 was 98.69 (b).
As the results, the thermal neutron flux was 1.94×10^4 $(n/cm^2/sec)$ and the calculated thermal neutron flux in the sub-pile room was 3.0×10^4 $(n/cm^2/sec)$ using PHITS2 [2] with JENDL-5 [3]. The experimental and calculated values agreed by an order of magnitude. In future work, we plan to perform an neutron flux evaluation without relaxes the above assumptions.

REFERENCES:
[1] T. Sano, *et al.,* KURNS-Progress report 2022, pp.71, (2023).
[2] T. Sato, *et al., J. Nucl. Sci. Technol.,* **55**, 684-690 (2018).
[3] O. Iwamoto, *et al., J. Nucl. Sci. Technol.,* **60**, 1-60 (2023).