CO3-1 Subcriticality measurement by using a small neutron detector (3)

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INTRODUCTION: In the fuel debris removal process of Fukushima Daiichi nuclear power plant, subcriticality monitoring system should be equipped to prevent criticality accident. For this purpose, International Research Institute for nuclear Decommissioning (IRID) is developing criticality control techniques for fuel debris removal based on neutron noise analysis using Feynman-alpha method or Rossi-alpha method. Prototype of the sub-criticality monitoring system was tested to verify applicability on various sub-criticality measurement conditions.
For this measurement, a small neutron detector based on a SiC with boron coated film is one of the candidates at Fukushima because of its toughness against gamma-ray and neutron radiation exposure and low detection efficiency for gamma-ray. We are also developing a data transfer system from this SiC neutron detector to data acquisition system which is placed at outside of a reactor vessel by a specially designed optical fiber with high resistance against radiation. In this research, we used this new data transfer system to measure subcriticality.

EXPERIMENTS: Experiment was carried out at KUCA solid moderated core, B-core, as shown in Fig.1 whose fuel assemblies were 3/8”p36EU with relatively soft neutron spectrum. The fuel coupon plates were sandwiched with polyethylene plates and assemblies were surrounded by polyethylene reflector to simulate water. This core was in subcritical state with low subcriticality with Cf-252 neutron source inserted in a fuel assembly. Data transfer system is illustrated in Fig.2. Boron-lined neutron detector was inserted in a periphery fuel region whose neutron detection analog signal was transfer pre-amplifier and then a data sender system by a co-axial cable. In the data sender system, analog signal was changed to optical digital signal and it was transferred to the data receiver system located at outside of the reactor room by a thin and long quartz optical fiber cable. Then digital data was changed to analog data in the data receiver system and finally neutron detection time whose time bin was 1 micro-second was transferred to PC by USB cable and stored in PC.

DATA ANALYSIS: Neutron detection time stamp data stored in PC whose time unit was 1 μs were analyzed by the neutron noise analysis methods, Feynman-alpha method and Rossi-alpha method.

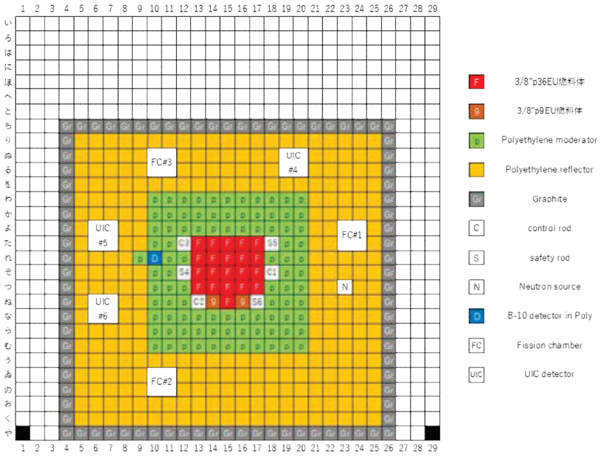

Fig.1 Core configuration.

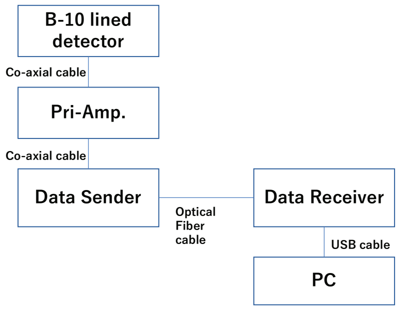

Fig.2 Data transfer system.

Table 1 Experimental results.

Subcriticality (%dk/k)	Calculated Alpha (s^{-1})	Experimental Alpha (s^{-1})
1.08	382	405±2
1.38	442	468±2
1.60	487	511±2
1.90	547	590±2
2.11	588	656±2

RESULTS: Results of Feynman-alpha method are shown in Table 1, where prompt neutron decay constants (alpha value) in various subcriticalities are listed. It is found that alpha value is close to calculated values in near critical state.
We continue to develop this data transfer system based of optical fiber combined with a SiC neutron detector for future usage.

CO3-2 Reactor Noise Power-Spectral Analysis for a Graphite-Moderated and -Reflected Core (III)

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INTRODUCTION: In the last study, a neutron detector located about 55 cm away of fuel assembly measured the auto power spectral density. However, the prompt neutron decay constants obtained by this detector was different from that of other detectors. The objective of this study is experimental study of reactor noise analysis by the power spectrum method using neutron detector placed outside reactor core.

EXPERIMENTS: The core configuration is shown in Fig. 1. “F” is test zone fuel assembly. “D” is a driver fuel assembly. “G” is graphite reflector. Yellow cell is polyethylene reflector. “1” to “6” are BF_3 proportional neutron counter positions. The dimensions these counters are 1.0 in. diameter and 15.47 in. length. The distance from core region to detector positions “1” and “2” are about 15cm, that to detector position “3” is about 20cm, that to detector position “4” is 35cm, that to detector position “5” is 55cm, that to detector position “6” is about 75cm.

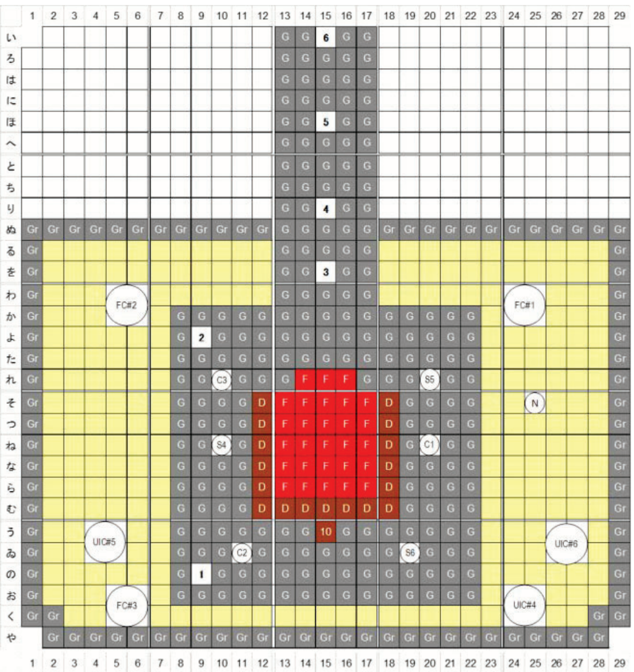

Fig. 1 Core configuration.
(B7/4”G2/8”p8EU(3)+3/8”p38EU)

Reactor noise analysis by power spectrum method was performed in the critical state of suitable reactor power for each detector position. The reactor power during measurement was adjusted so that the count rate of each detector was about 3,000[cps].

RESULTS: The auto power spectral densities and cross power densities by detector position ”3” to “6” are shown in Fig. 2 and Fig, 3. These figures also include least squares fits of a conventional formula [1] to the spectral densities to determine the prompt neutron decay constant α_0 (β_{eff}/Λ), where the fitting was confined to a frequency range from 2.5 to 100 Hz or 2.5 to 50 Hz. The prompt neutron decay constant obtained by the detectors “3” and “4” were the same within the error range. However, the decay constant obtained by detector “5” and “6” were very different. In the future, it is necessary to improve the fitting formula and reexamine the analysis frequency range of power spectrum method.

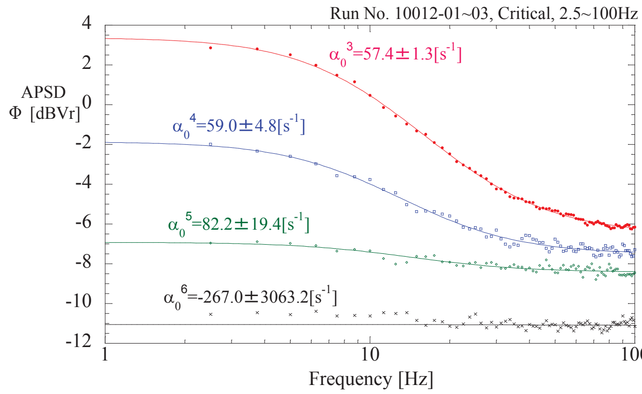

Fig. 2 Auto power spectral densities.
(Detector “3” to “6”)

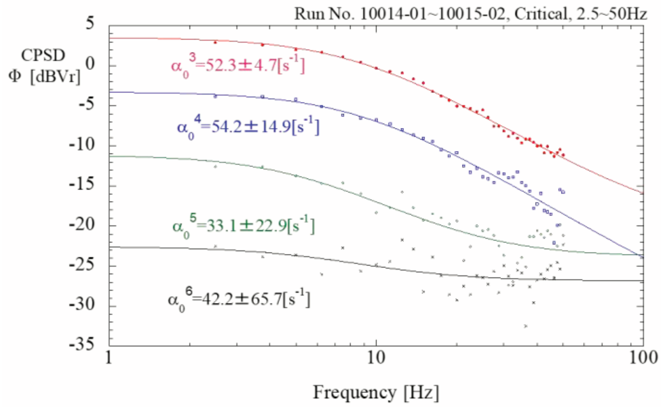

Fig. 3 Cross power spectral densities.
(Detector position “3” to “6”)

REFERENCES:

[1] M. M. R. Williams, Random Processes in Nuclear Reactors, (Pergamon Press, Oxford,1974), section 3.6.

R2CA07

CO3-3 Reactor Physics Experiment in a Graphite Moderation System for HTGR (III)

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INTRODUCTION: The Japan Atomic Energy Agency (JAEA) started the Research and Development (R&D) to improve nuclear prediction techniques for High Temperature Gas-cooled Reactors (HTGRs) in 2018. The objectives are to introduce the generalized bias factor method to avoid full mock-up experiment for the first commercial HTGR and to improve neutron instrumentation system by virtue of the particular characteristics due to a graphite moderation system. For this end, we composed B7/4"G2/8"p8EU(3)+3/8"p38EU in the B-rack of Kyoto University Critical Assembly (KUCA) in 2021.

EXPERIMENTS: The core configuration is shown in Fig.1. The objective is to cover the characteristics of commercial HTGR with relatively high enriched fuel. The Gas Turbine High Temperature Reactor 300 (GTHTR300) employs 14wt% enriched uranium fuel to achieve high burn-up of 120GWd/t. The core of B7/4"G2/8"p8EUNU+3/8"p38EU composed in 2018 has the fuel assembly averaged enrichment of which is 5.41wt% to realize a characteristics of High Temperature Engineering Test Reactor (HTTR). The characteristics of GTHTR300 are expected to be evaluated by the interpolation of the characteristics of the two cores from the viewpoint of generalized bias factor method because the present core is composed of only highly enriched uranium plate.

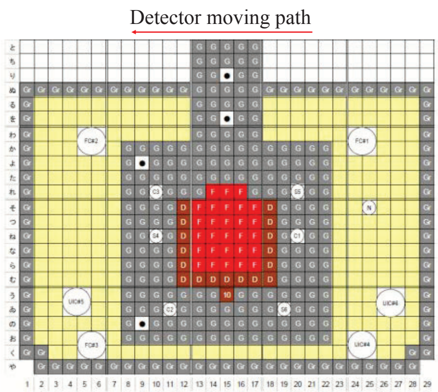

Fig. 1 Core configuration.

To demonstrate the power distribution measurement system from ex-core detector which have been developed by JAEA[1], not only a noise analysis, which was performed by the detector expressed by a dot mark in Fig. 1, but also neutron detector signals measurement by a moving detector as shown in Fig. 2 were performed.

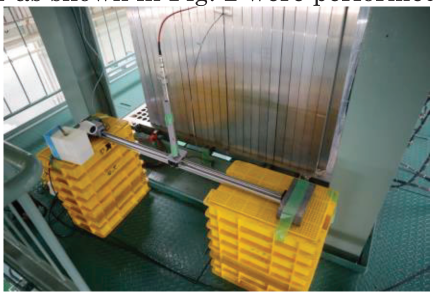

Fig.2 Moving detector system.

RESULTS: The spectrum of the core is compared with that of GTHTR300 in Fig.3. The Maxwell peak is successfully realized and close sensitivity is expected.

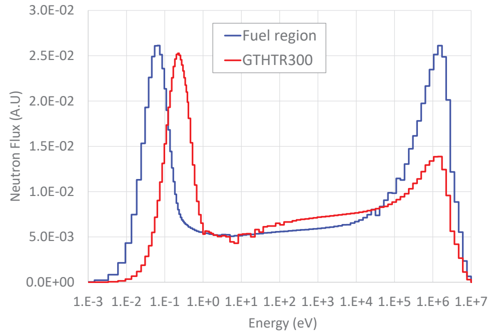

Fig.3 Neutron spectrum.

The moving detector system is expected to demonstrate a power distribution measurement from ex-core detector. By virtue of a long neutron flight path of a graphite moderation system, the power distribution can be evaluated by inverse analysis with measurement data by moving detector system. The feasibility was already proved by a numerical simulation in HTTR geometry [1]. The detector signals along with the moving path illustrated in Fig.1 are shown in Fig. 4. The detector moves 1m stroke by 87 sec. The expected detector signals were obtained compared with calculation result by MVP code.

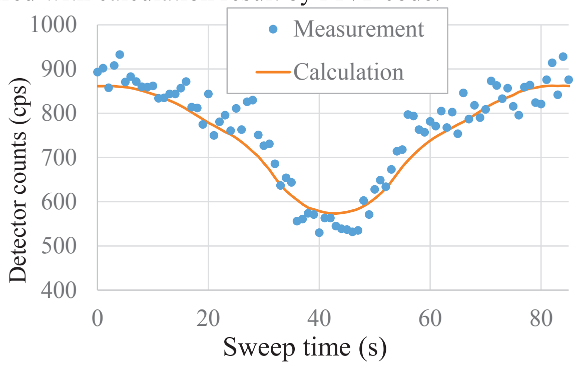

Fig.4 Neutron detector signals

Reference:

[1] Y. Fukaya, S. Okita, S. Nakagawa, *et al.*, "Computed tomography neutron detector system to observe power distribution in a core with long neutron flight path," Ann. Nucl. Enegy 168, pp.108911_1-108911_7,(2022).

CO3-4 Measurement of fundamental characteristics of nuclear reactor at KUCA (VII)

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ACKNOWLEDGEMENTS: The authors This work is a part of "Development of Safety System (Liquid/Gas System, Technologies for Criticality Control" project supported by the Ministry of Economy, Trade and Industry (METI). The authors would like to express their deep thanks to Mrs. S. Wada2, 3, M. Shimizu2, 3 and H. Yazawa2, 3 for their helps in performing the KUCA experiment.

INTRODUCTION: The reactor noise methods can measure the subcriticality without additional instruments. International Research Institute for Nuclear Decommissioning (IRID) is developing the monitoring subcriticality system based on the reactor noise methods using the Feynman-alpha method. A prototype system was developed using B-10 neutron detectors [1]. The system is required with radiation hardness. We, IRID, picked up the corona detectors and multi-cell detector as the candidate detectors with radiation hardness.
For this measurement, the applicability of the monitoring subcriticality systems with corona counters with B-10 or He-3 and a multi-cell He-3 detector are verified.

EXPERIMENTS: Experiment was carried out at KUCA solid moderated core, B-core, as shown in Fig.1 whose fuel assembly was 3/8"p32EU. We measured some cores that changed number of the fuel assemblies, in order to simulate shallow/deep subcriticality and situation of high neutron count-rate. This core was aimed to compare the porotype system with B-10 detectors used in past experiments. In Fig.1, D1 is mean a corona He-3 detector, D2 is mean a corona B-10 detector, and D3 is mean a multi-cell He-3 detector.

RESULTS: As shown in Fig. 2, the testing systems can measure the available time-list data for the Feynman-alpha method. The comparison of k-effective evaluated by time-list data of each system is shown in Fig. 3. These results those k-effective are higher than 0.95 are good agreement with reference k-effective simulated by the MVP code with JENDL-4. The results in the deep subcriticality less than 0.7 of k-effective are agreement with 10% error.
The test results showed that the testing systems are feasible to measure the subcriticality.

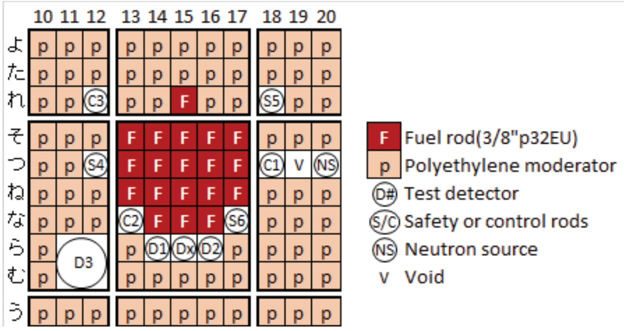

Fig. 1 Core configuration.

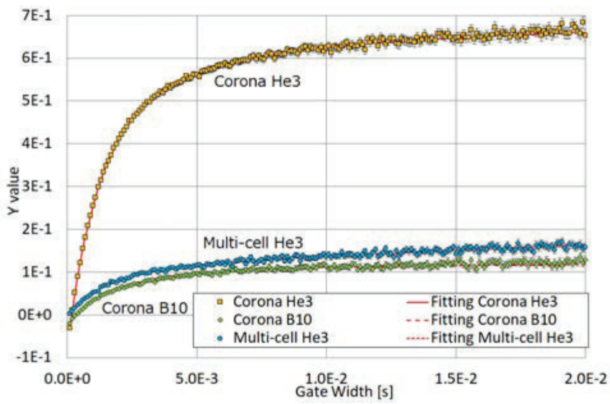

Fig. 2 Example of Feynman-Y value & curve.

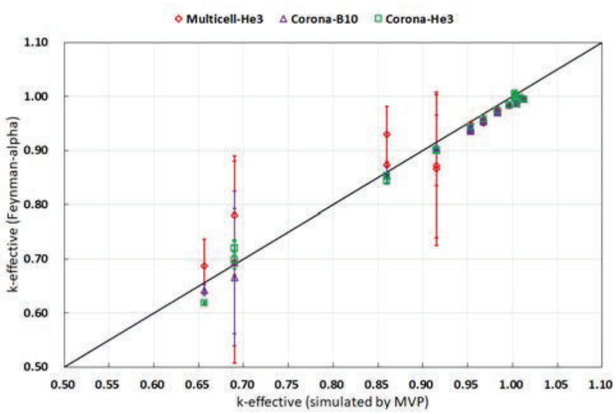

Fig. 3 Comparison of k-effective.

REFERENCES:
[1] S. Wada *et al.*, *Proc. of ICNC2019*, Paris France, Sep. 15-20 (2019).

CO3-5 Verification of a method to estimate reactivity in a deep subcritical state

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INTRODUCTION: The estimation of reactivity of an amount of unknown fissile material is one of important issues in the field of criticality safety.
JAEA has been theoretically developing a method to estimate the reactivity from neutron count rate alone[1-2]. The method is based on a newly developed equation of power in quasi-steady state after prompt jump/drop of power due to reactivity and/or neutron source change.
The purpose of the experiment is to obtain the experimental data for the verification and validation of the developed method. This time, the data were obtained for deep subcritical states around keff = 0.95, which is a threshold value for a subcritical condition in a numerical analysis.

EXPERIMENTS: A subcritical experiment was done with the basic critical core configuration known as 3/8" p36EU of A-core. The Am-Be was used as the external neutron source.
^3He detectors were used. Figure 1 shows the core configuration and the position of Am-Be neutron source.

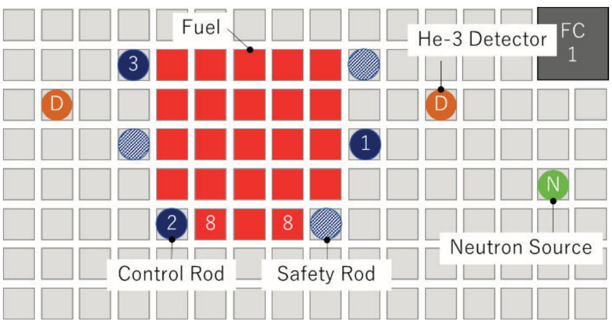

Fig. 1. Configuration of fuels and devices in A-core.

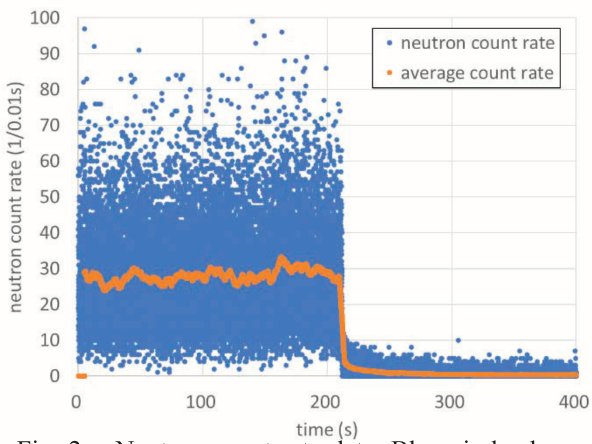

Fig. 2. Neutron count rate data. Blue circle shows neutron counts per 0.01s and orange circle shows a profile of averaged one.

For the first several hundred seconds, as shown in Fig.2, the system was kept under a subcritical steady state. Then a negative reactivity was inserted by dropping control rods, C1-C3, safety rods, S4-S6, and/or Center Core. The achieved keff is shown in Table1. 6 rods were dropped for 0.97, Center Core for 0.95 and the rods and CC for 0.93. Those keff values were roughly estimated by summing up the reactivity worth of each device.
After that, neutron count rate decreased and the measurement was terminated after several hundred seconds or more.

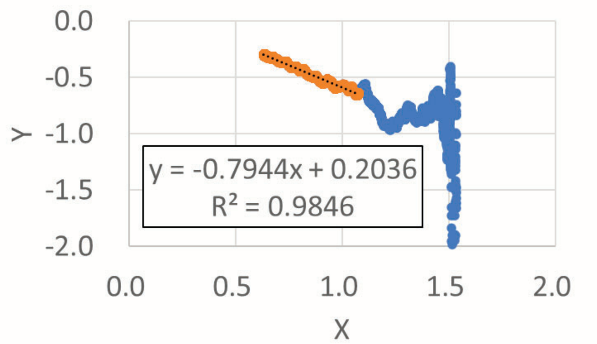

Fig. 3. Linear fitting to the points (X, Y) calculated from the neutron count rate data shown in Fig. 2.

RESULTS: Reactivity value was estimated following the procedure described in [2], in which the neutron count rate data from the detector far from the neutron source was chosen and treated. The points $(X(t), Y(t))$ were calculated from the neutron count rate data and its slope value, $a = -0.794$, was estimated by linear fitting as shown in Fig. 3. Then the reactivity value was calculated as follows;

$$\rho_\$ = \frac{1}{\left({}^{1}\!/_{a} + 1\right)} = -3.9$$

The estimated value obtained by preparatory calculation with the new method was summarized in Table 1. The difference between the reference value and estimated one is up to 12%. It shows the applicability of the new method to deep subcritical state, and more detailed analysis is needed to quantify its degree of accuracy.

Table 1. The experimental condition and estimated reactivity value by applying the new method.

operation	keff	reactivity		
		reference ($)	estimated($)	difference(%)
drop C1-S6	0.97	-4.1	-3.9	-5.4
drop CC	0.95	-8.1	-9	12
drop all	0.93	-12	-12	-5

REFERENCES:
[1] Y. Yamane, J. Nucl. Sci. Technol. Vol.57(8), 926-931 (2020).
[2] Y. Yamane, J. Nucl. Sci. Technol. https://doi.org/10.1080/00223131.2022.2053221 (2022).

CO3-6 Basic Experiment on Reactor Power Distribution Reconstruction by Ex-core Neutron Detectors

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INTRODUCTION: We are developing an in-core power distribution estimation method (PHOEBE) using ex-core neutron detectors to reduce the cost and improve the maintainability of nuclear instrumentation in small reactors as a distributed power source [1]. Theoretical and numerical examinations were initially studied with the experimental demonstration conducted at UTR-KINKI [2]. The demonstration was conducted with a simple core Hence, more complex experiment geometry is required to evaluate the power distribution of the core inner region for the experimental demonstration of PHOEBE. In this study, a new core was constructed at KUCA and tested to confirm the effectiveness of PHOEBE.

EXPERIMENTAL GEOMETRY: Figure 1 shows a configuration of fuel element in this experiment. A unit cell consisted of one 1/16" EU plate, two 1/8" PE plate and one 1/4" graphite plate. The fuel element had 30- or 14-unit cells and graphite reflectors. Figure 2 shows the experimental core layer. 31 fuel elements were loaded into the core and 3 control rods, 3 safety rods, graphite reflectors elements were installed. In addition, 8 eight neutron detector systems (ERYNGII), which can efficiently measure epi-thermal neutrons as shown in Fig. 3, were inserted for this experiment.

RESULTS: Table 1 shows the measured neutronics characteristics. The all characteristics were satisfied with the KUCA regulations. In this core, the power distribution was distorted by control rod manipulation or special fuel element, and the epi-thermal neutron flux were measured by ERINGII. As a result, the validity of PHOEBE was confirmed.

REFERENCES:
[1] R. Kimura, *et. al.*, Nucl. Sci. Eng. 2020; 195(12), pp1279-1290.
[2] R. Kimura, *et. al.*, AESJ 2021 Fall Meeting, 2I09, (2021).

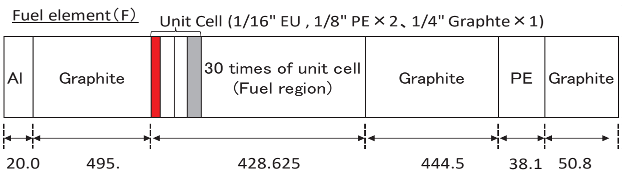

Figure 1 Configuration of fuel element.

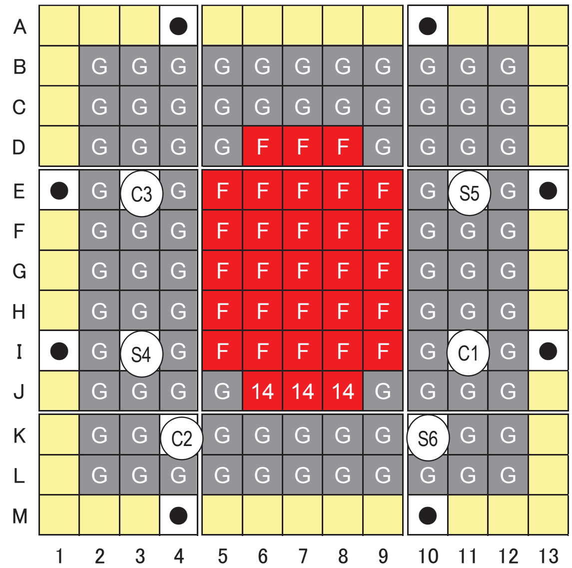

Figure 2 Experimental core.

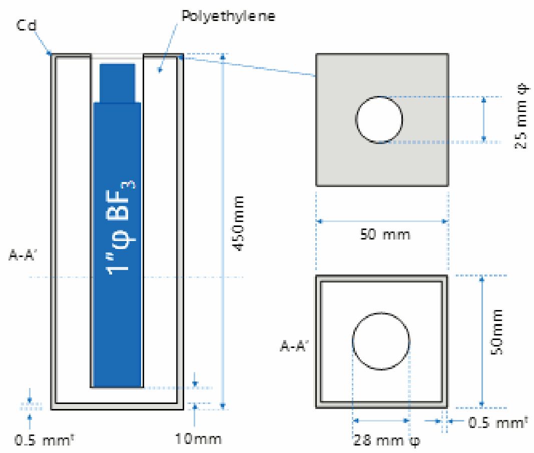

Figure 3 Schematic diagram of neutron detector system. (ERYNGII)

Table 1 Measured neutronics characteristics.

Neutronics Characteristics		Measured	KUCA Regulation
Excess reactivity (%dk/k)		0.154	< 0.35
Max. reactivity insertion rate (%dk/k/sec)		0.007 C3:830mm-730mm	< 0.02
Rod worth (%dk/k)	C1	0.599	Max. worth < 1/3 of total worth
	C2	0.422	
	C3	0.654	
	Total	(0.599+0.422+0.654)x2=3.350	Total worth > Excess + 1 %dk/k
Center core worth (%dk/k)		3.325	> 1%dk/k

CO3-7 Measurement of Neutronics Characteristics for Th loaed core at KUCA (III)

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INTRODUCTION:
In engineering discussions of the feasibility of new reactor systems, it is necessary to evaluate the impact of the fuels and materials for the neutronics characteristics such as criticality, conversion rate, and fuel balance. In order to develop of Thorium (Th) nuclear system, critical experiments on Th loaded cores using the KUCA with solid moderator core have been systematically carried out to perform neutronics characteristics measurements of Th loaded thermal neutron systems and integral evaluation of Th cross sections [1]. In order to perform nuclear design for Th loaded reactors, it is important to validate U-233 nuclear data. In order to perform an integral validation of U-233 fission cross section, measurements of sample reactivity worth in KUCA solid moderated core were carried out.

EXPERIMENTS:
In this experiment, two types of fuel element were loaded into the KUCA solid moderated core. Figure 1 shows composition of a fuel element "F" and a special fuel element "S". The fuel element "F" was consisted of 31 unit cells and sandwiched by a upper and a lower polyethylene reflector. The unit cell has one enriched uranium plates of 0.159 mm (1/16 in.) thickness, three polyethylene plates of 0.318 mm (1/8 in.) thickness. The fuel element "S" was consisted of 30 unit cells and one sample case made of aluminum. The geometry of sample case has 50.8 mm × 50.8 mm × 11.11 mmt (7/16 in.).

A sample plate was made of U_3O_8-Al and the amount of U-233 in a plate was about 0.09 g. The U-233 enrichment was 99.4 wt% [2]. The geometry of a plate was 12.66 mm × 12.68 mm × 1.08 mm^t and has a round chamfering (R = 3 mm). 9 plates of U-233 sample were set into a aluminum sample case. Figure 2 shows a core configuration of this experiment. The experimental core has 24 fuel elements "F" and one special fuel element "S". The C1 – C3 are control rods and the S4 – S6 are safety rods.

RESULTS:
The sample reactivity value was defined as the difference of excess reactivity between the core with the aluminum case installed the U-233 sample plates and the core with the aluminum case without the samples. The excess reactivity measurements were carried out by the positive period method. In each measurement, the C2 and C3 control rods were set to the upper limit (U.L.) and the C1 control rod was adjusted to the critical position. And the C1 control rod was drawn out to the U.L. from the critical position and the reactivity was inserted into the core. As the result, the excess reactivity with and without U-233 sample were 0.1003 ± 0.0032 %dk/k and 0.0649 ± 0.0014 %dk/k [2]. Therefore, the sample worth was obtained as 0.0354 ± 0.0015 %dk/k [2].

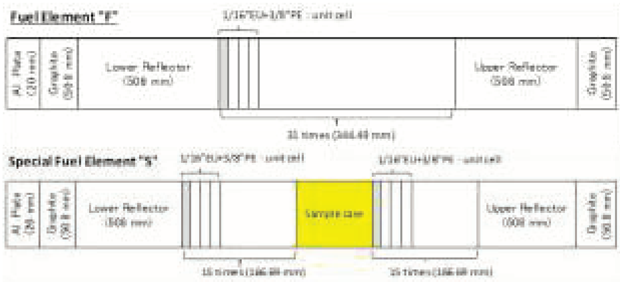

Figure 1. Configuration of a fuel element "F" and "S".

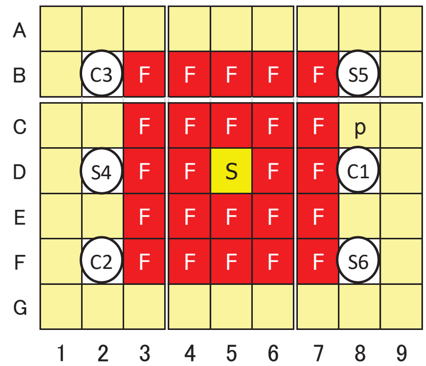

Figure 2 Core configuration.

ARKNOWLEGEMENT:
A part of the present study was supported by CHUBU Electric Power Co., Inc.

REFERENCES:
[1] T. Sano, *et. al.*, Proc on PHYSOR2020, Cambridge, UK, Mar.29th – April 2nd, (2020).
[2] T. Sano, *et. al.*, Abstract of AESJ 2021 Fall meeting, Online, 1F14, (2021).

CO3-8 Subcriticality measurement by using a small neutron detector (4)

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INTRODUCTION: Subcriticality monitoring system has to be used to detect criticality approach for each step of debris removal in Fukushima Daiichi nuclear power plant. For this purpose, International Research Institute for nuclear Decommissioning (IRID) is developing criticality control techniques for fuel debris removal based on neutron noise analysis using Feynman-alpha method. A prototype of the sub-criticality monitoring system was tested to verify applicability on various sub-criticality measurement conditions.
For this measurement, a small neutron detector based on a SiC with boron coated film is one of the candidates at Fukushima because of its toughness against gamma-ray and neutron radiation exposure and low detection efficiency for gamma-ray. We are also developing a data transfer system from this SiC neutron detector to data acquisition system which is placed at outside of a reactor vessel by a specially designed optical fiber with high resistance against radiation. In this research, we used this new data transfer system to confirm the availability for subcriticality measurement.

EXPERIMENTS: Experiment was carried out at KUCA solid moderated core, B-core whose fuel assemblies were 3/8"p36EU. Data transfer system is illustrated in Fig.1. Boron-lined neutron detector was inserted in a periphery fuel region whose neutron detection analog signal was transfer pre-amplifier and then a data sender system by a co-axial cable. In the data sender system, analog signal was changed to optical digital signal and it was transferred to the data receiver system located at outside of the reactor room by a thin and long quartz optical fiber cable. Then digital data was changed to analog data in the data receiver system and finally neutron detection time whose time bin was 1 micro-second was transferred to PC by USB cable and stored in PC.
In the previous year experiments, unexpected noise signal sometimes appeared, for example once in 5 minute, in accumulated data, which cause inaccurate subcriticality results. This unexpected data transfer may be caused from data transfer system, however, we cannot clear up this phenomena up to the present time.
To remove those unexpected noise signal, we divided time stamp data into several part, and each time period data was analyzed by the neutron noise analysis methods, Feynman-alpha method and Rossi-alpha method. Then, we compared those results to confirm existence of noise signal.

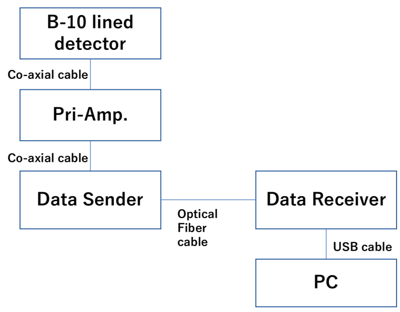

Fig.1 Data transfer system.

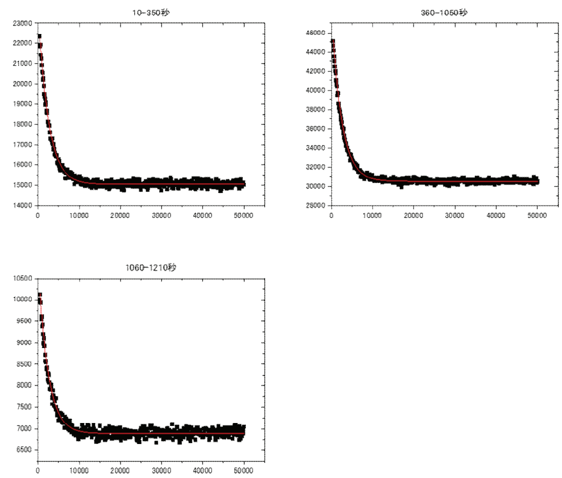

Fig.2 Rossi-alpha results by dividing time stamp data.

RESULTS: The results are shown in Fig. 2 for Rossi-alpha method after removing unexpected time stamp data. By using this data dividing technique, it is possible to remove unexpected noise signal data.
We continue to improve this data transfer system based of optical fiber combined with a SiC neutron detector for future usage.

CO3-9 Measurement of Validation Data for Kinetics Parameter and Reactor Kinetics

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INTRODUCTION: Precise estimation of the reactor kinetics is essential for the nuclear safety. In CRIEPI, continuous energy Monte Carlo (MC) method has been studied to estimate the point kinetics parameters in a critical condition [1, 2] and reactor periods [3]. Besides, time dependent neutron transport calculation techniques are now under development [4, 5]. In order to validate those calculations, comprehensive data sets of reactor kinetics were measured in A3/8"p36EU(3) core.

RESPONSE FOR STEP REACTIVITY INSERTION: Positive and negative reactivities were inserted step wisely by moving C1 rod and time variation of power was measured as shown in Fig. 1 (left). The reactivity range was from -46 to 51 pcm. The asymptotic period given by the flux a few hundred seconds after the insertion shall be used to validate whether the effective delayed neutron fraction repartition per precursor family, $\beta_{eff,i}$, relates the measured periods to the calculated static reactivities [1,2]. The transient phase from the prompt jump to the asymptotic behavior is also useful to validate time dependent calculation. For that, the prompt jump behavior was focused by raising the initial critical power (Fig. 1, right).

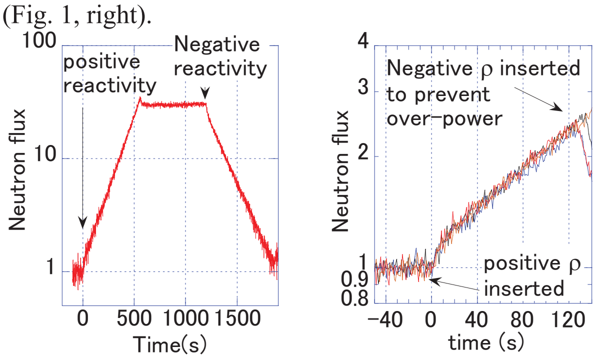

Fig.1 Response to positive & negative step reactivity insertion (left). Focusing on prompt jump (right).

EFFECT OF REACTIVITY INSERTION RATE: Reactivity insertion rate is also important for kinetics behavior of a core. In this work, we attained criticality by adjusting the C1 rod position (688 mm) while C2 and C3 rods were withdrawn. Then the C1 rod was fully inserted from time 0 second by 1) dropping and by 2) continuous down with speed of -8.013 mm/s. Neutron flux decrement with time was measured three times for both the insertion conditions. The results are shown in Fig. 2. In the rod drop measurements, the negative reactivity was measured -404, -392, -394 pcm for the case 1, 2, 3, respectively. The time dependent neutron flux of case 2 and 3 are similar and that of case 1 is bit different. Due to the instant insertion of the negative reactivity, negative prompt jump was observed. Whereas in the continuous down case, the time dependent neutron flux of the three cases is similar each other and that is very credible as the validation data. According to the worth curve of C1 rod and the insertion speed, the negative reactivity of -392 pcm is given within 34.5 s. After 35 s, the flux by the continuous down of C1 rod approaches to that by the rod drop test. These data shall be applied to validate the time dependent MC calculation.

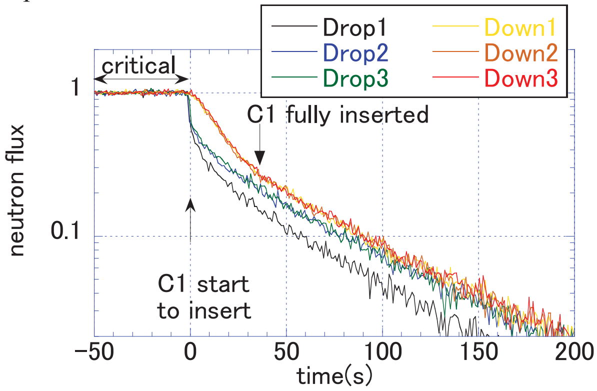

Fig. 2 Neutron flux decrement by C1 rod insertion.

VARIANCE TO MEAN METHOD: For the validation of the generation time Λ, measurement of α eigenvalue is useful. For that, the Feynman - α measurement was done with a conventional method. The neutron count rate in FC #2 and #3 were measured with dwell time of 0.1 ms and the Y value and α were preliminary evaluated without correction for dead time of detectors. β_{eff} and Λ are calculated with enhanced MCNP-5.1.30 with the AcelibJ40 library. The reactivity ρ is given based on the rod worth curve based on $\beta_{eff,ij}$ by MCNP calculation [1,2]. Then the calculated α is given as $\alpha = (\beta_{eff} - \rho) / \Lambda$. The measured and calculated αs are compared in Table 1. The C/E ranges from 0.985 to 1.126. We shall investigate the influences of the dead time and spatial higher-mode flux to make C/E more credible.

Table 1: Measured and calculated α eigenvlaue.

Reactivity (pcm)	Measured $\alpha (s^{-1})$ by FC#2	C/E	Measured $\alpha (s^{-1})$ by FC#3	C/E
-329	214.09	1.040	225.92	0.985
-144	171.44	1.084	174.46	1.065
-72	161.23	1.065	152.93	1.122
0	139.64	1.126	155.08	1.014

ACKNOWLEDGEMENT: The authors are so honored to have been a part of the historical moment when the Run No. of KUCA reaches 10,000 since its commencement in 1974.

REFERENCES:
[1] Y. Nauchi *et al.*, *JNST* **47**(11)977-990, 2010.
[2] A. Zoia *et al.*, *ANE* **96** 377-388, 2016.
[3] Y. Nauchi *et al.*, ANE162 108486, 2021.
[4] Y. Nauchi, *JNST* **56**(11)996-1005, 2019.
[5] Y. Nauchi, Proc. M&C2021, Oct.3-7, 2021.

CO3-10 Basic Research for Sophistication of High-power Reactor Noise Analysis (IV)

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INTRODUCTION: Reactor noise for high-power reactors were actively measured in the 1960's and 1970's. The major focuses of those researches were for the abnormality diagnosis or the output stabilization diagnosis, and almost researchers were in the field of system control engineering or instrumentation engineering. High-power reactor noise measurements for dynamics' analysis of reactivity change, reactivity feedback or reactor characteristics itself were few in the time (1960's and 1970's), because of the powerless measurement system. In this research, we plan to measure KUR's output with present-day measurement system and plan to analyze with several analysis methods. The results of this work will supply some knowledges and technics in the aspect of sophistication of reactor noise analysis or simulation methods.
In this year, we tried to measure the reactor nuclide noise of the critical state KUR core via a 1-inch ^{3}He counters at CN-1 port focused on epi-thermal neutrons. The experimental work was done on 25th November 2021. As the result of the experiment, a result looks like the nuclear reactor noise was observed in 1MW critical state.

EXPERIMENTS:
In this experiment, the output signal of the ^{3}He counters (LND 25291×3) were put into Spectro Scopy AMPs (2022: Canberra and 590A: ORTEC), and the output of the SSAs were measured with a time-series measurement system (HSMCA4106_LC: ANSeeN Inc.). A schematic view of the measurement is shown in Fig.1, and the counter installation overview is shown in Pic.1.
As you can find in Pic.1, the neutrons from the KUR core were measured with ^{3}He counters via a 5-mm thickness B_4C sheet.

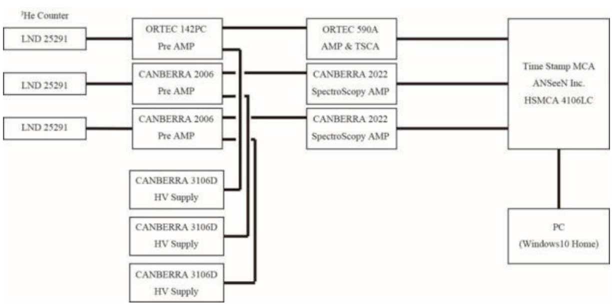
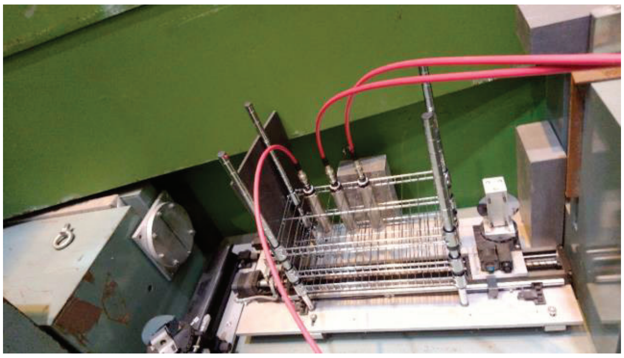


Fig. 1. Schematic view of the measurement.

The experimental condition is shown in Table.1. The reactor Power was set in 1MW. The measurement time was 8,000 sec.



Pic. 1. An overview of the counter installation

Table 1. Experimental condition

Reactor Power [W]	Measurement Time [sec]	Count Rate [cps]
1M	8,000	620 - 660 (#1) 210 - 230 (#2) 560 - 600 (#3)

RESULTS:
The measurement results were analyzed by Feynman-α / bunching method, Rossi-α method and Covariance to Mean Ratio method.
As a result of the Feynman-α analysis, plot shapes like Feynman's theoretical formula were not obtained, because of the gradually increasing of the KUR power.
As a result of the Rossi-α analysis, plot shapes like Orndoff's theoretical formula were obtained on the 3 counters' results. As a result of the Covariance to mean ratio analysis, plot shapes like Feynman's theoretical formula were obtained, however the Y values were extremely small.
An analysis result example of the Rossi-α analysis is shown in Fig.3. The result of this work show that the neutron energy selection may improve the sensitivity of the reactor nuclide-noise measurements for the light water moderation reactor.

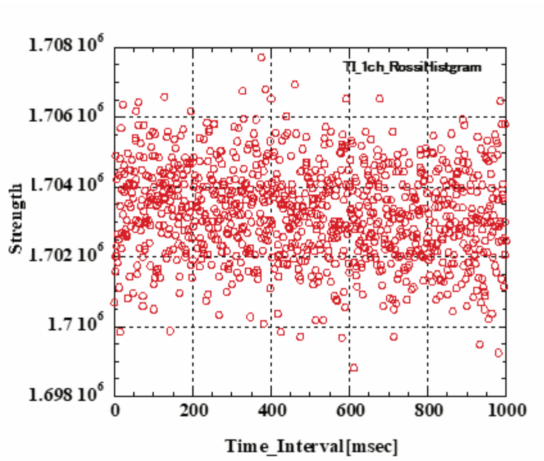


Fig. 2. A result example of Rossi-α analysis

CO3-11 Development of Faster Measurement Method of High Neutron Flux with SPND

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INTRODUCTION: To safely control the reactor, monitoring neutron flux inside the reactor is necessary. Therefore, a reliable technology is needed to measure neutrons of high intensity, as in the reactor. Self-powered neutron detectors (SPNDs) are available as detectors to perform the measurement. The SPND is possible to obtain information on neutron flux by measuring the current signal generated with the use of activation of emitter material (Rh, V, Co etc.). In the detection principle, the response time of the detector is determined by the time constant of activation decay, which makes it difficult to respond quickly. In this study, we will explore a method to speed up the response of SPND by digital data processing.

EXPERIMENTS: Since the self-output neutron detector requires high neutron intensity, neutron measurements were conducted in the inclined irradiation borehole (SLY) of the Kyoto University research Reactor (KUR), which allows measurements to be made close to the reactor core in this test. The preliminary measurement was performed with the detector inserted during 1 MW operation, and the next measurements of output power were carried out at the KUR operation ranging between 1 and 5 MW. The detector was a self-output neutron detector using rhodium (Rh) as the detector. The output of the self-output neutron detector was measured using a pico-ampare-meter.

RESULTS: Figure 1 plots the current from the SPND versus time versus the linear output meter of the furnace; the output of the SPND at 1 MW output was about 5 nA the response time of the detector after insertion into the SLY was about 300 s. The response time is due to the decay constants of the activation products of ^{113}Rh, ^{114}Rh and ^{114m}Rh, of 40 s and 4.4 min. When the output was changed from 1 MW to 5 MW, the maximum current value from the SPND was 24 nA at 5 MW.

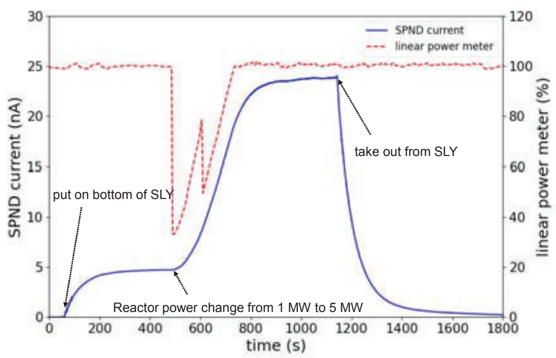

Fig. 1 Plot of SPND measured current and linear poser meter ratio via time.

To accurately examine the time response of the SPND, we examined the time variation after the detector was pulled out from the SLY. The results of exponential fitting to the time variation are shown in Figure 2. Figure 2 indicates that the time variation is confirmed to be reproduced by adding the two exponential curves together. The time constants of the response were 41 s and 4.3 min, respectively. The ratio of the current values derived from ^{104}Rh and ^{104m}Rh was 1:0.07. Using the parameters, we will investigate a useful way to estimate the output of SPNDs more quickly.

In this study, we considered the method to estimate the output from the activation equation. The activation equation can be expressed by Eqs. (1) and (2) as follows:

$$N_{Rh^{104}} = -\lambda_{Rh^{104}} N_{Rh^{104}} + \lambda_{Rh^{104m}} N_{Rh^{104m}} + N_{Rh^{103}} \sigma f \quad . \tag{1}$$

$$N_{Rh^{104}} = -\lambda_{Rh^{104m}} N_{Rh^{104m}} + N_{Rh^{103}} \sigma f \quad , \tag{2}$$

where σ is the activation cross section, f the neutron flux, and N the number of atoms.

The two equations are different from each other, and the filtered neutron fluxes are plotted in Figure 3. The filtered result were found to respond faster that the raw measurement result.

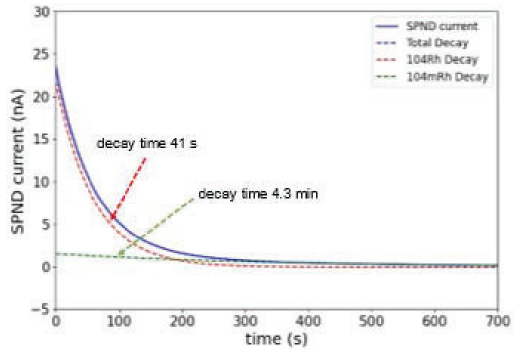

Fig. 2 Plot focused on SPND current decay after the detector pulled out from SLY. The decay curve was fitted with the two exponential curves.

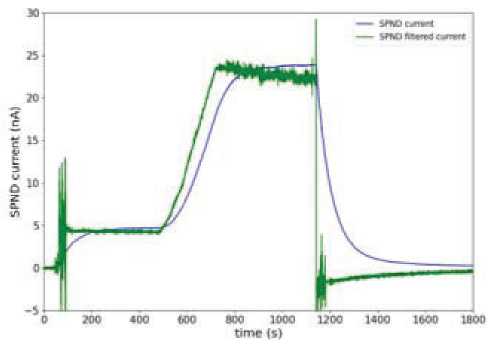

Fig. 3 Comparison between the raw SPND current and the filtered one.