CO3-1 Development of In-Reactor Observation System Using Cherenkov Light (IV)

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INTRODUCTION: In research reactors, CCD cameras are used to observe reactor core for reactor operation management, e.g. to prevent debris from falling. In order to measure the reactor power and fuel burnup exactly by means of observation of Cherenkov light [1], the development of the on-line measurement device started in 2009. In this study, the wavelength and the absolute irradiance of the Cherenkov light were measured by a spectrometer, and the Cherenkov light was observed by the CCD camera.

EXPERIMENTS: The in-reactor surveillance system was composed of the spectroscopy system and a visible imaging system. The spectroscopy system (Maya2000, OptoSirius Corp.) was able to measure the wavelength of light in the range from 200 nm to 1100 nm. The visible imaging system was composed of the CCD camera (AEC-100ZL, Q・ I Inc.), a monitor recorder, and a controller. The latter system was used to observe the Cherenkov light with changing the aperture value. For in-reactor surveillance, the spectrometer and the CCD camera were inserted into core-observation pipe of KUR. The Cherenkov light was measured when the reactor power steadily increased from 0.2 MW to 1 MW, and was at 5MW. The ND-filter and the aperture were changed during the observation in order to confirm their performance for the Cherenkov light.

RESULTS: The absolute irradiance of the Cherenkov light as a function of wavelength is shown in fig. 1. The result shows that the detected Cherenkov light, which has a maximum value of absolute irradiance at 400 nm, distributes from 380 to 700 nm in wavelength independent of the reactor power. The relationship between the nominal transmittance value of ND-filter and the measurement value is shown in fig.2. The result shows the measurement value by the spectroscopy system was well accorded with the nominal transmittance value of ND-filters at the reactor power of 0.5, 1, and 5MW. As the result of observation of Cherenkov light by the visible image system, the images were fogged by halation during high power operation. However, the amount of incident light could be controlled by changing the ND-filters, and halation could be removed from those images.

CONCLUSION: The Cherenkov light from KUR core was measured with the developed in-reactor surveillance system. The correlation between absolute irradiance of the Cherenkov light and the nominal transmittance value of ND-filters was evaluated by the spectroscope system. As a result, the measurement value is good agreement with the nominal value. On the other hand, the value by the visible imaging system was obtained the same tendency of nominal transmittance value of ND-filters. These results suggest that a possibility of a device to obtain the reactor power and fuel burnup information by observing the Cherenkov light.

REFERENCES:

[1] J.V. Jellry, Cherenkov Radiation and its Applications (Pergamon, New York, 1958).
[2] N. Takemoto, K. Tsuchiya, *et al.*, KURRI Progress Report, (2010) P. 204.

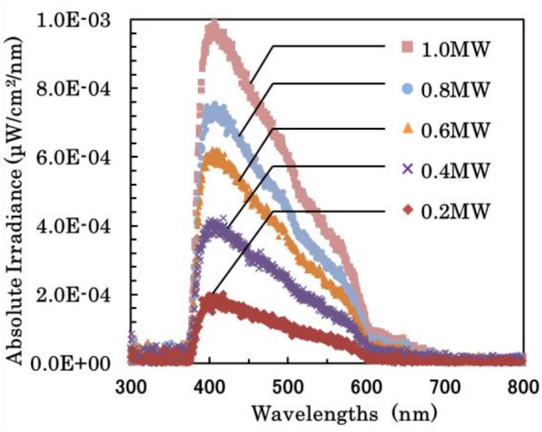

Fig. 1. The absolute irradiance of the Cherenkov light as a function of wavelength.

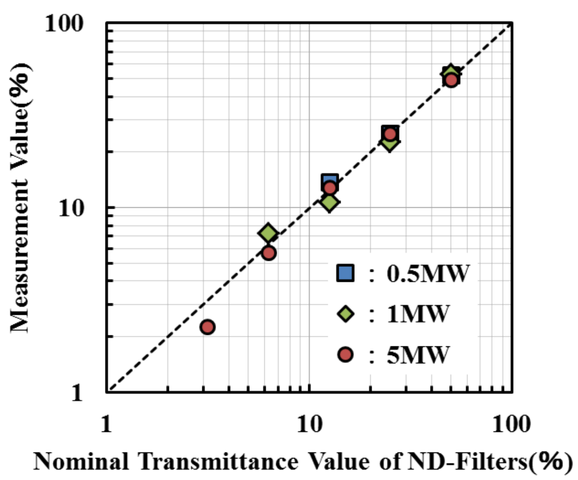

Fig. 2. Characteristics of the ratio of transmitted light of the Cherenkov light by ND-filter.

採択課題番号 24038　チェレンコフ光を用いた試験研究炉の炉内監視手法の研究開発　　共同通常
（原子力機構）土谷邦彦、谷本政隆、那珂道裕、竹本紀之、柴田　晃、武内伴照、木村明博、
西方香緒里、木村伸明（京大・原子炉）中島　健、宇根崎博信、佐野忠史、藤原靖幸、奥村　清

CO3-2

Evaluation of Activation Products for Decomissioning

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INTRODUCTION: In the case of decommissioning, radioactive inventory is used as a basic data for planning of decommissioning, occupational exposure estimation, estimation of radioactive waste and public safety assessment. Then, evaluation of radioactive inventory is necessary information to carry out decommissioning safely and reasonably. According to previous studies [1], accuracy improvement of BSW concrete (the largest amount of material in NPP) radioactive inventory evaluation has been already considered. But, for the second largest amount of material, metal, accuracy improvement is insufficient. This study is focused on improving accuracy of radioactive inventory evaluation of metals, considering the difference of neutron flux [2].

EXPERIMENTS: Before experiment, activation calculation was conducted with two elemental composition data; JAERI-M [3] and chemical analysis date of sample.

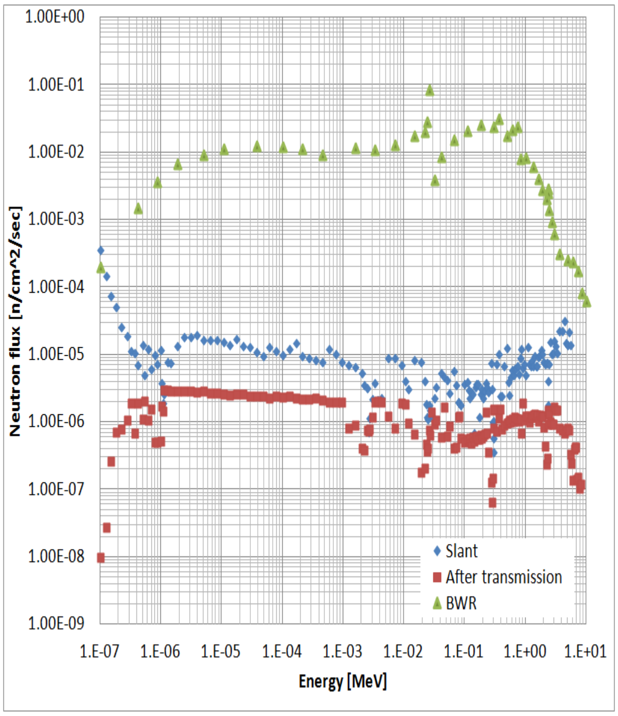

Fig. 1. Neutron energy spectrum of inner surface of biological shield concrete.

In neutron irradiation experiment, samples are same standard metals used in practical commercial reactor. But the neutron spectrum in slant is different from a typical LWR. In order to reproduce the neutron spectrum in a commercial LWR, an optimal moderator should be selected. The transmission spectrum is calculated by using MCNP-5 (Monte Carlo Simulation Code). Adjustment of neutron spectrum is shown in Fig.1. For experiment, Cd is selected as moderator. After irradiation experiment, sample is measured by Ge semiconductor detector.

RESULTS: The result of SUS304 is shown in Table 1. Measurements and calculated values showed a good agreement, but there is a difference of 10^3 times between calculation data and measured data in ^{60}Co.

Table 1. Experiment and calculation data of SUS304

Nuclide	Calculated data [Bq/g]		Measured data [Bq/g] (RSD)
	JAERI-M 6928	chemical analysis	
^{51}Cr	9.93E+03	9.12E+03	5.49E+03(±2%)
^{58}Co	6.88E+02	7.87E+02	3.86E+03(±4%)
^{60}Co	2.04E-01	1.02E+02	1.04E+02(±9%)
^{54}Mn	9.06E+01	9.77E+01	5.17E+01(±12%)
^{56}Mn	5.56E+06	5.51E+06	3.76E+06(±1%)
^{59}Fe	1.28E+02	1.42E+02	9.80E+01(±15%)

REFERENCES:

[1] T. Ogawa, *et al.*, Radiation Protection Dosimetry 146(1-3), p.356-9 (2011)

[2] N. Kawata, *et al.*, JNST, Vol. 9, No. 4, p. 405-418 (2010)

[3] K. Koyama, *et al.*, JAERI-M 6928 (1977)

採択課題番24070　　放射性核種生成量評価のための中性子による材料照射の研究　　一般通常
（東大院・工）小佐古敏荘、飯本武志、谷幸太郎、藤通有希、熊谷一城、小坂晃義、渡邉貴裕、矢埜孝三

CO3-3

Performance Evaluation of Radionuclide Monitoring Systems during Reactor Operation

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INTRODUCTION:

Examinations of two kinds of radionuclide monitoring systems using coincidence counting were conducted in the Kyoto University Research Reactor (KUR). One uses a reactor water monitor and the other uses a ^{60}Co monitor for the primary loop recirculation (PLR) piping.

It is important to detect fuel rod failure early in order to improve safety of nuclear reactors. One of the measures for early detection is changing from sampling surveys to continuous monitoring, which is defined as online monitoring. Some isotopes of radioiodine can be used as index radionuclides to ensure fuel integrity.

In addition, in order to reduce dose exposure caused by ^{60}Co which is the main radionuclide encountered by workers during scheduled outages of boiling water reactors, water chemistry control during reactor operation is carried out[1]. It is necessary to monitor the deposited ^{60}Co on the inner surfaces of PLR piping to evaluate effects of water chemistry control.

However, many short-half-life radionuclides like ^{13}N, ^{16}N and ^{19}O present in the reactor water interfere with the detection of gamma rays emitted by the iodine isotopes and ^{60}Co. ^{132}I, ^{134}I and ^{60}Co have the characteristic of emitting gamma rays in a cascade. By using a coincidence method, the gamma rays emitted by these radionuclides can be measured with a high signal-to-noise ratio because most interfering radionuclides emit gamma rays independently. Thus, monitoring systems based on the coincidence counting method were developed.

EXPERIMENTS:

(1) Reactor water monitor

Two $LaBr_3$(Ce) scintillation detectors were set up near the ion exchanger in the KUR. A standard mixed source was also set up 2.5 to 3.0 cm from the detectors. The source included two radionuclides emitting gamma rays in a cascade, ^{60}Co (1012 Bq) and ^{88}Y (430 Bq). The air dose rate around the detectors was estimated with a pocket dosimeter and counting rates were as high as 40 μSv/h at the time of the experiments. The gamma rays were detected and their pulse-height and time information was recorded with the measurement system. The information was used to measure the cascade radionuclides by the coincidence method in an offline process.

(2) ^{60}Co monitor for PLR piping

Two $LaBr_3$(Ce) scintillation detectors and a ^{60}Co source, which was used to simulate the ^{60}Co deposited on the inner surfaces of PLR piping, were set up in the heat exchanger room of the KUR; this arrangement could simulate the radiation environment inside of a primary containment vessel (PCV). The air dose rate during reactor operation was about 4.8mSv/h. Experimental measurement instruments using the coincidence counting method were set up in the cold area.

3. RESULTS:

(1) Reactor water monitor

As shown in Fig. 1, due to accidental coincidence, net coincidence counting rates decreased with increasing air dose rate. By converting ^{60}Co measurement data to ^{132}I and ^{134}I for the reactor water conditions, it was found that more than 100 min were needed to detect fuel failure with the device and a signal-to-noise ratio more than 10 times larger was needed to satisfy the measurement limit defined by regulation. Thus, the measurement efficiency of the system must be enhanced by 10 times.

(2) ^{60}Co monitor for PLR piping

The transition of coincidence counts rate for ^{60}Co is shown in Fig. 2. The dotted line was calculated based on the geometry of the two detectors and the ^{60}Co source, the detection efficiency of the gamma rays and the ^{60}Co source activity. It was confirmed that the coincidence counts for ^{60}Co could be detected for the dose rate of 4.8 mSv/h with a standard deviation of 19%.

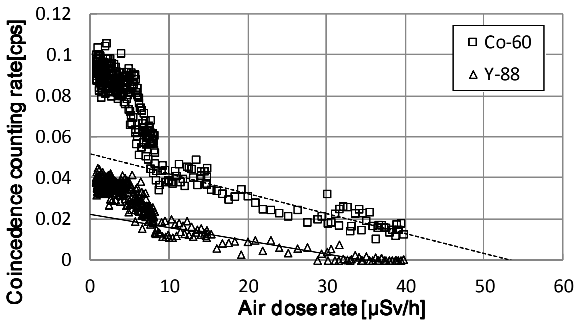

Fig. 1 Net coincidence counting rate.

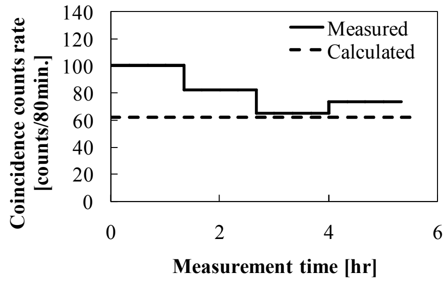

Fig. 2 The transition of coincidence counts for ^{60}Co.

REFERENCE:

[1] "Research Report Related to the Trend of the Water Chemistry of the Domestic BWR Plants", Japan Nuclear Energy Safety Organization, Sep. 2012, pp.10-15 (in Japanese).

採択課題番号 24093 高バックグラウンド条件における Co-60 及び原子炉水中核種の検出　共同即時
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（日立）田所　孝広、上野　克宜、岡田　耕一

CO3-4 Neutronic Characteristics of Lead in KUCA A core for Accelerator-Driven System

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INTRODUCTION: An Accelerator-Driven System (ADS) has been investigated in Japan Atomic Energy Agency (JAEA) to transmute minor actinides discharged from nuclear power plants. The ADS proposed by JAEA is a lead bismuth eutectic (LBE) cooled-tank-type ADS. It has been known that there was a major upgrade for the cross section data of lead isotopes from JENDL-3.3 to JENDL-4.0 and the upgrade affects to the neutronic design of the ADS [1, 2]. For instance, k_{eff} value calculated by JENDL-3.3 was 0.97 and the value calculated by JENDL-4.0 was 1.00 and the main cause of this difference was the cross section data of the lead isotopes[2].

This study aims to measure replacement reactivity from aluminum plates to lead plates to validate the nuclear data of the lead isotopes.

EXPERIMENTS: The experiment was carried out in the KUCA A core (EE1) shown in Fig. 1. A special assembly which included Al or Pb plates instead of polyethylene was placed at the center of the core, そ-15. For the special assembly, five different loading patterns were employed by changing the number of Al and Pb plates as shown in Fig. 2. Case Al60 was the reference case and the difference of excess reactivities between Al60 and other cases were calculated as a replacement reactivity. Case Pb40C was the case where 40 Pb cells were placed in the center of the fuel assembly and case Pb40E was the case where 40 Pb cells were put in the edge of the fuel assembly 20-and-20.

Figure 3 presents the experimental results for each case. As the number of the Pb plate increased, the replacement reactivity from Al to Pb also increased. The replacement reactivity was 20.8, 44.7 and 48.6 pcm in the case of Pb40C, Pb40E and Pb60, respectively.

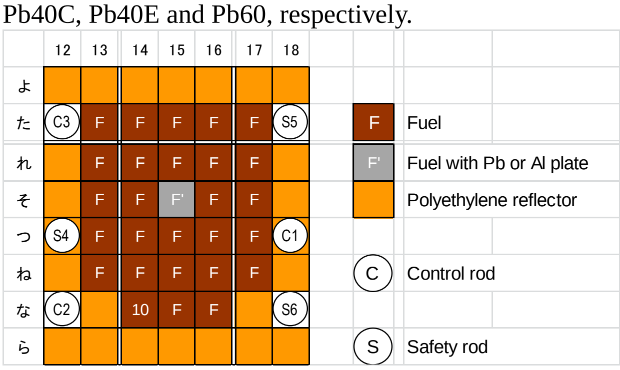

Fig. 1. Core arrangement in the KUCA-A core (EE1)

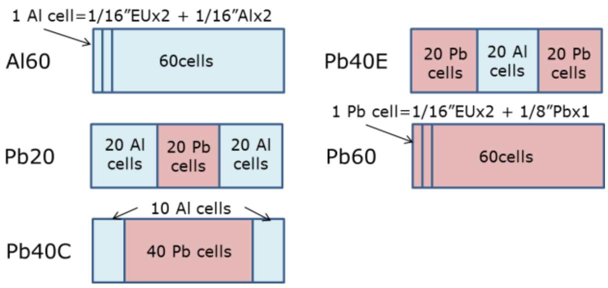

Fig. 2. Fuel cell for each case

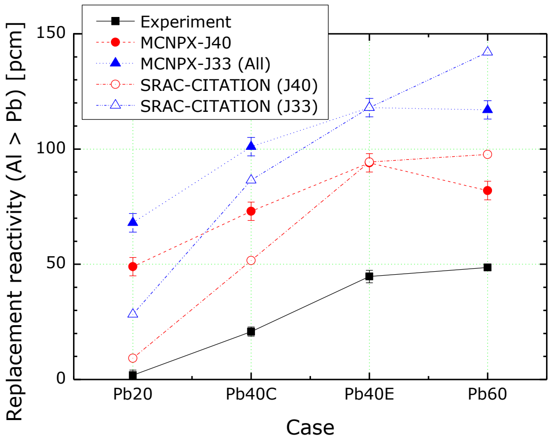

Fig. 3. Comparison of experimental results and calculation results

CALCULATIONS: MCNPX and SRAC-CITATION codes were used for the calculation with JENDL-4.0 and JENDL-3.3 libraries. The k_{eff} value was calculated for each experimental case and the difference of the k_{eff} value was calculated as the replacement reactivity.

Figure 3 also shows the calculation results for each case. For the MCNPX calculation, although all calculation results were larger than the experimental ones, the results calculated with JENDL-4.0 were close to the experimental ones. The same tendency was observed in the SRAC-CITATION calculation. It was guessed that the nuclear data of the lead isotopes in JENDL-4.0 was more reasonable than those in JENDL-3.3. On the other hand, it is necessary to investigate the large discrepancy between the experiment and calculation results.

CONCLUSION: The replacement reactivity from the aluminum plates to the lead ones was measured and the calculation results indicated that the lead nuclear data in JENDL-4.0 might be more reasonable than those in JENDL-3.3.

REFERENCES:
[1] T. Sugawara *et al.*, Atom Indonesia, 38, 2, (2012) 71-77.
[2] H. Iwamoto, *et al.*, Proc. ND2013, New York, Mar. 4-8, (2013)

採択課題番号 CA24101　KUCA-A 架台を用いた加速器駆動炉における鉛の核特性評価　共同通常
（日本原子力研究開発機構）菅原隆徳、西原健司、岩元大樹（京大・原子炉）卞哲浩、八木貴弘

CO3-5 Development of Activation Foil Neutron Detector for Angular Distribution Measurements of Epi-Thrermal Neutrons in Reactors

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INTRODUCTION: For basic design and performance predictions of radiation shielding, numerical simulations are quite important. Although the accuracy of the numerical simulations on the neutron transportation is reliable, basic experiments still play an important role especially for neutron deep penetration problems. The validity of the numerical simulations has been confirmed through comparisons with the integral experiments, because it is difficult to experimentally obtain differential information, such as neutron energy and directions. So far, we have developed the measurement system of the angular distribution of thermal neutrons. This system can successfully observe the neutron streaming effect between the solid moderators of the Kyoto University Critical Assembly (KUCA). As the next step, we develop a new directional detector for epi-thermal neutrons. This system applies an activation foil with resonance neutron absorption in the epi-thermal energy region as a neutron sensitive element. In this report, we demonstrate the direction distribution measurements performed at the KUCA.

ACTIVATION FOIL DETECTOR FOR ANGULAR DISTRIBUTION MEASUREMENTS: As the activation foil for detection of the epi-thermal neutrons, we adopt a silver foil. Silver-109 with the isotopic abundance of 48% has a large resonance absorption peak at 5.2 eV. Figure 1 shows the schematic drawing of the proposed angular distribution measurement system for the epi-thermal neutrons. We fabricated the prototype detector system. The fabricated system consists of an irradiation capsule covered with a silver activation foil, an irradiation part with a silver collimator for confining the incident direction of epi-thermal neutrons, a plastic scintillator for detection of β particles emitted from the activated silver foil and a pneumatic carrier system for the activation capsule transportation. In order to measure the directional distribution, the irradiation capsule is cyclically transported between inside and outside a reactor with a pneumatic system and the irradiation part can rotate.

We demonstrated the directional distribution measurements of epi-thermal neutrons at the solid moderator core (A core) in the KUCA. The plastic scintillator for β-ray detection was placed outside the reactor and surrounded with a lead shielding to suppress background radiations. Figure 2 shows the directional distribution of the detected counts by the fabricated detector system. The irradiation part was placed on the center line of the core and shifted to right side. The directional responses corresponding to the detector positions were confirmed to be observed. We concluded that the fabricated detector system can acquire the information of the directional distribution of epi-thermal neutrons.

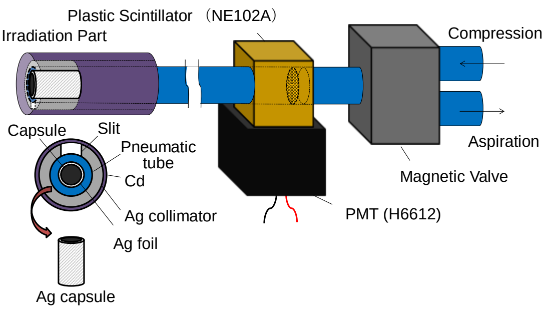

Fig. 1 Schematic drawing of the angular distribution measurement system for epi-thermal neutrons.

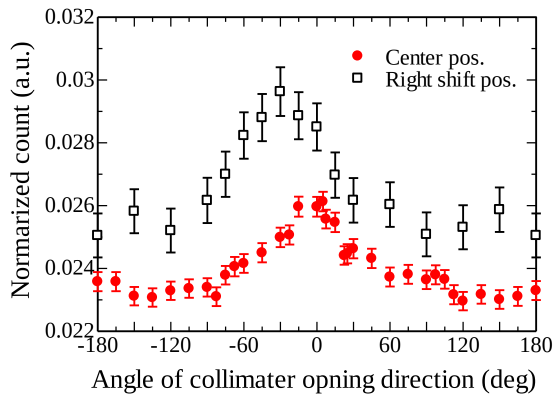

Fig. 2 Directional distribution of the detected counts by the fabricated detector. The irradiation part was placed on the center line of the core and shifted to right side.

採択課題番号 CA24103 性子場特性評価を目的とした新型中性子検出器の開発に関する研究 共同通常
（名大・工）渡辺賢一、瓜谷章、山﨑淳、丸山秀典、川端勇矢、牛田雅人
（京大・原子炉）三澤毅、卞哲浩

CO3-6 Development of Subcriticality Measurement for Accelerator-Driven Reactor (VII)

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INTRODUCTION: A subcritical reactor system driven by pulsed spallation neutrons generated from tungsten or lead-bismuth target has been constructed in A-loading facility of the KUCA and a series of cross-power spectral analysis between time-sequence signal data from two neutron detectors has been performed to develop the methodology of on-line subcriticality monitoring for future accelerator-driven system. The preliminary results of the cross-power spectral analyses are showed in this report.

EXPERIMENTS: The present analyses were carried out in a reactor system referred to as A1/8"P60EU-EU. The tungsten or lead-bismuth target was placed outside polyethylene reflector and the spallation neutrons were generated through the injection of 100MeV protons onto the tungsten or lead-bismuth target. As a pulse repetition frequency (period), 20Hz (50ms) was employed. The signals from three fission counters (FCs) for reactor startup operation were fed to a fast Fourier transformer to analyze the cross-power spectral density and to record the signals as digital data. An analysis range in frequency from 1.25 to 1000Hz was specified to obtain 800-point spectral data from the transformer. From the spectral density, the prompt-neutron decay constant was determined. As a subcritical rod pattern for this experiment, we employed a shutdown pattern where all control and safety rods were completely inserted.

RESULTS: Figure 1 shows a measured cross-power spectral density between two fission counters FC1 and FC3. We can observe many delta-function-like peaks and this feature is similar with that observed in a previous analysis for 14MeV source [1]. However, no continuous reactor-noise component can be seen from this figure. This is because these fission counters are located far from the core and consequently the efficiency is extremely low.

Assuming the one-point kinetics model, the top points of the uncorrelated peaks can be described as

$$\Phi(\omega) = A_0 \sum_{m=1}^{\infty} \frac{\delta(\omega-\omega_m)}{\alpha_0^2+\omega^2}, \tag{1}$$

where $$\omega_m = 2\pi m / T_R . \tag{2}$$

In the above equations, ω is angular frequency, α_0 the prompt-neutron decay constant of fundamental mode and T_R pulse repetition period. A_0 is a constant dependent on detection efficiency, generation time, and accelerator parameters. The least-squares fits of equation (1) to the peak point data for tungsten and lead-bismuth target lead to the prompt-neutron decay constants of 1076.8±33.9 and 1060.2±32.5[s^{-1}], respectively. These fits underestimate the decay constant, compared with 1226.1±5.3[s^{-1}] obtained by a pulsed neutron experiment. Figure 1 also shows a significant deviation of the fitted curve from the peak points in the higher frequency range over 500Hz.

Considering the contribution of a spatially higher mode, the top points of the uncorrelated peaks can be rewritten as

$$\Phi(\omega) = A_0 \sum_{m=1}^{\infty} \frac{\delta(\omega-\omega_m)}{\alpha_0^2+\omega^2} + A_1 \sum_{m=1}^{\infty} \frac{\delta(\omega-\omega_m)}{\alpha_1^2+\omega^2}. \tag{3}$$

A further least-squares fit of the above equation (3) to the point data of delta-function-like peaks is shown in Fig.2, where the deviation observed above disappears. The least-squares fits of equation (3) for tungsten and lead-bismuth target lead to the consistent prompt-neutron decay constants of 1208.9±50.2 and 1206.3±60.2[s^{-1}], respectively.

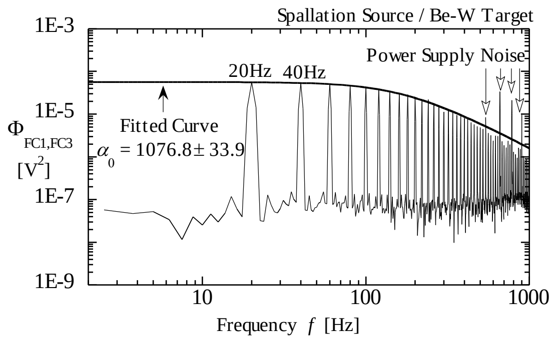

Fig.1 Fit to peak points of equation (1) to uncorrelated peak points

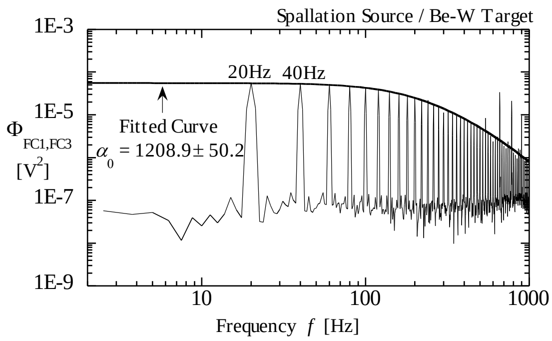

Fig.2 Fit to peak points of equation (3) to uncorrelated peak points

REFERENCES:

[1] A. Sakon *et al.*, J. Nucl. Sci. Technol., **50** (2013) 481-492.

採択課題番号 CA21404 加速器駆動未臨界炉における 共同通常
未臨界度測定高度化のための基礎実験（Ⅶ）
（近大・原研）橋本 憲吾、杉山 亘（近大院・理工）左近 敦士、ムハマド・アイマン・ビンマーロフ、河崎 将大（京大・原子炉）卞 哲浩、佐野 忠史、八木 貴宏、三澤 毅

CO3-7 Active Gamma Ray Spectroscopy of Sub-Critical System Containing Stainless Steel

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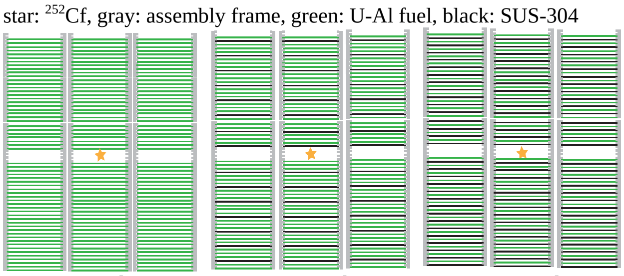
INTRODUCTION: In units 1 - 3 of the Fukushima Daiichi power station, it is anticipated that molten fuel has been mixed with reactor structural materials, which act as a neutron absorber. In order to certify the criticality margin brought by the mixture of the structural materials, reaction rates of the neutron capture by them should be quantified in relation to fission reaction rates. For the purpose, sub-critical experiments using uranium – stainless steel mixed cores have been launched for the first time.

EXPERIMENTS: SUS-304 (Fe, Ni (8-10.5%), Cr (18-20%)) is a typical structural material. SUS-304 plates (600mm x 62mm x 1.5mm) were prepared together with the U-Al alloy fuel plates (93% ^{235}U enrichment) and 3 types of core were mocked up as shown in Fig. 1. Mass ratios of SUS-304 to Uranium in the cores are 0, 15.4, and 46.3, respectively. The subcritical multiplication factors of the cores are 0.86, 0.71, and 0.52, respectively, when a ^{252}Cf neutron source are loaded at the center of the cores. Thermal neutron flux distributions on the outer surface of the cores were measured with the ^{6}Li scintillators. Gamma ray spectra were measured with a BGO scintillator at 46 cm outside from the cores.

star: ^{252}Cf, gray: assembly frame, green: U-Al fuel, black: SUS-304



SUS-304: 0 plates
Fuel: 180 plates

SUS-304: 45 plates
Fuel: 135 plates

SUS-304:90 plates
Fuel: 90 plates

Fig.1 SUS-304 loaded cores.

RESULTS: The measured neutron flux distributions agree well with the calculations by the MCNP-5 code with the nuclear data libraries based on the JENDL-4 as shown in Fig. 2. This means that the neutronics calculations have enough accuracy even in the sub-critical systems containing stainless steel (SUS-304).We calculated the reaction rates of the neutron capture by SUS-304 and fission of ^{235}U. Then the capture to fission ratio (C/F) was deduced for each core.

The gamma ray pulse height spectra for the three cores measured with the BGO scintillator are shown in Fig. 3. The gamma rays from capture reactions of ^{56}Fe, ^{27}Al, and ^{58}Ni are observed in 6-10MeV region and continuous spectra mainly caused by the fission reactions are found in 3-5MeV region. As the SUS-304 loading ratio increases, the count rate in the 6-10MeV region increases and that in the 3-5MeV region decreases. By coupled neutron-photon transport calculations, the detection efficiencies of the gamma rays were deduced. Using the detection efficiency and the measured count rates, gamma ray emission rates were deduced. The ratio of gamma ray emitted in 6-10MeV to 3-5MeV energy regions are plotted against the calculated C/F in Fig. 4. The gamma ray emission ratio monotonously increases with C/F. Accordingly, it was confirmed that the neutron capture rate by the structural materials can be quantified in relation to the fission rate based on the active gamma ray spectroscopy.

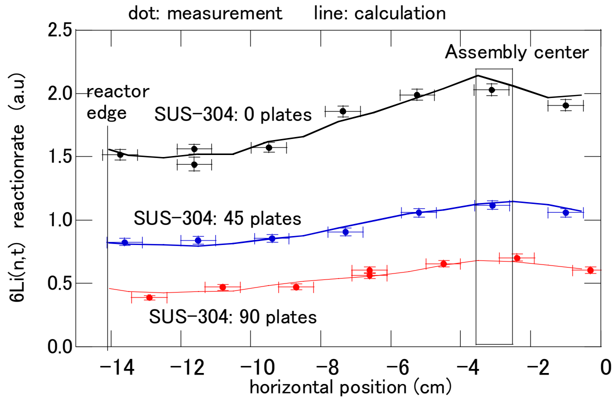


Fig.2 Thermal flux distribution outside cores

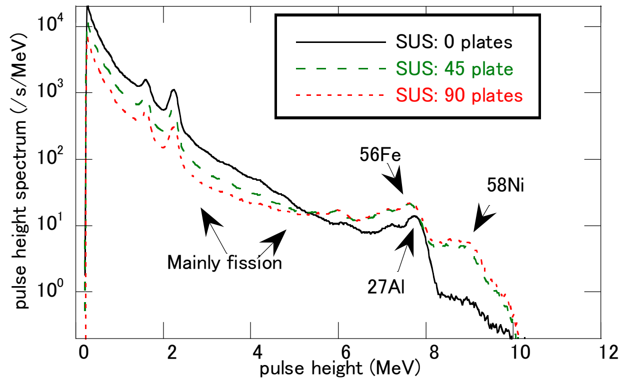


Fig.3 gamma ray pulse height spectra for sub-critical cores measured by BGO scintillator.

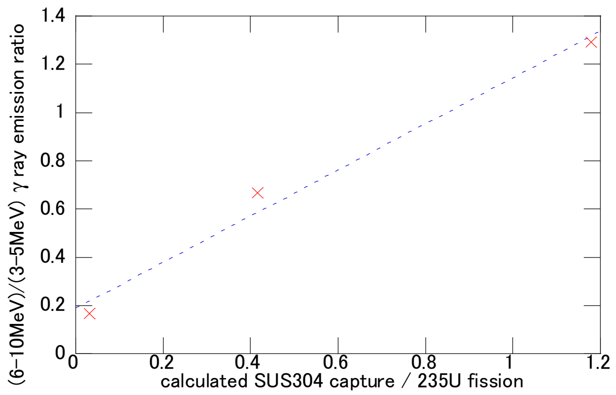


Fig.4 (6-10MeV) / (3-5MeV) gamma ray emission ratio against calculated capture (SUS-304) / fission (^{235}U) one.

採択課題番号 CA24106 γ線計測を利用した未臨界度指標の計測に関する研究(2) 共同通常
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CO3-8 Subcriticality Measurement Experiment Using Inherent Neutron Source

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INTRODUCTION: The Feynman-α method is a practical subcriticality measurement technique based on reactor noise analysis, since it can be carried out simply with an ordinary neutron detector system [1]. It is noted that the sufficient number of detected neutrons is required to accurately measure reactor noise. Thus, an external neutron source, e.g. Am-Be or Cf, is used to increase detected neutron counts. However, it is often unfavorable to perform measurement using an external neutron source in actual nuclear fuel facilities. Therefore, the subcriticality measurement using the inherent neutron source in nuclear fuels, such as spontaneous fission and (α,n) reaction, was demonstrated in the previous study [2]. The purpose of present study is to clarify the intensity of the inherent neutron source and the measurement time required for the noise analysis.

METHODOLOGY: In the Feynman-α method, the time series data of detected neutrons $C(T)$ within a counting gate width T is measured. Then, the neutron correlation factor $Y(T)$ is evaluated as follows:

$$Y(T) \equiv \sigma^2(T) / \mu(T) - 1, \qquad (1)$$

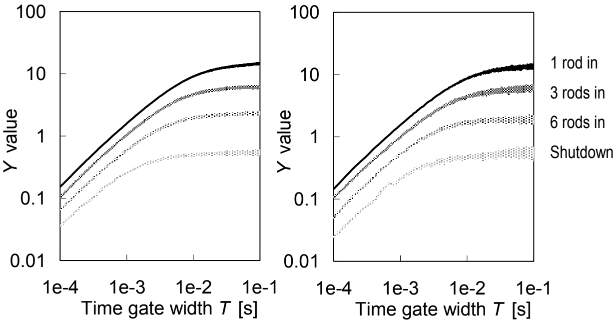
where $\mu(T)$ and $\sigma^2(T)$ is the mean and the variance of $C(T)$, respectively. As shown in Eq. (1), $Y(T)$ means a difference of "variance-to-mean ratio of $C(T)$" from unity, thus $Y(T) = 1$ if the frequency distribution of $C(T)$ is subjected to the Poisson distribution. On the basis of the one point reactor kinetic equation, theoretical expressions of $Y(T)$ and subcriticality $(-\rho)$ can be described as follows:

$$\mathrm{Y}(T) = \mathrm{Y}_\infty \left(1 - \frac{1 - e^{-\alpha T}}{\alpha T}\right), \quad (2); \qquad \frac{(-\rho)}{\beta_{\mathrm{eff}}} = \left(\frac{\alpha}{\alpha_0} - 1\right), \quad (3)$$

where Y_∞ is the saturation value of $Y(T)$; α and α_0 is the prompt neutron decay constant for target and critical systems, respectively; and β_{eff} is the effective delayed neutron fraction. Using the Feynman-α method, α can be obtained by the least square fitting of Eq. (2) to measured $Y(T)$. Finally, subcriticality in dollar unit, $(-\rho)/\beta_{\mathrm{eff}}$, can be obtained by substituting α into Eq. (3), when the value of α_0 is estimated by another measurement or numerical analysis.

EXPERIMENTS: The reactor noise experiments were carried out in A3/8"P36EU-NU(3) core without external neutron source. Four subcritical states, which are achieved by different patterns of six control rods and withdrawing fuel and reflector assemblies in shutdown, were measured. Four ^{3}He detectors were placed in outer reflector region to collect time series data of detected neutron counts. In the fuel plate used in KUCA, the inherent neutron sources mainly consist of spontaneous fission of ^{238}U and (α,n) reaction of ^{27}Al. The inherent neutron source intensity is ~1000 [neutrons/sec/core], which is estimated by the PHITS code [3] with TENDL-2011 alpha sub libraries. In order to verify the experiments only with inherent source, we also carried out the reactor noise experiments with external neutron source (Cf source in the center of core). In addition, the reference values of $(-\rho)$ were evaluated by the control rod worth and the excess reactivity, which were obtained by the rod drop and the period methods, respectively.

RESULTS: The values of $Y(T)$ were analyzed with the bunching method [4] using the time series data (1 and 10 min) measured only with the inherent source (Fig. 1). Figure 1 shows the saturation value Y_∞ is larger and the saturation time is longer, as the value of $(-\rho)$ close to zero. Figure 2 shows α values obtained by the least square fitting of Eq. (2) to measured $Y(T)$. As shown in Fig. 2, the 1- minute measurement only with inherent source is not sufficient to accurately measure α value, since the results of $Y(T)$ have relatively large statistical errors as shown in Fig. 1. By taking longer time series data, the α values obtained from 10-minutes time series data is sufficient in this experiment. Hence, it is confirmed that the α value can be measured by a reactor noise measurement only with the inherent neutron source, if the measurement time is sufficiently long.



(a) 10 min (b) 1 min

Fig. 1 Variations of $Y(T)$ for 4 subcritical states

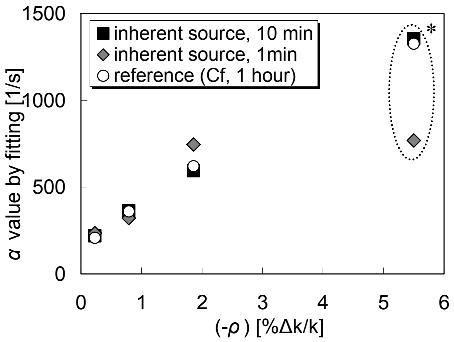


Fig. 2 α values plotted for $(-\rho)$

*$(-\rho)$ in shutdown state was estimated by Eq.(3) with α_0

REFERENCES:

[1] D. G. Cacuci, Handbook of Nuclear Engineering:Vol. 3 Reactor Analysis, Springer, 1644-1647 (2010).

[2] K. Tonoike *et al.*, *J. Nucl. Sci. Technol.*, **41**, 172-182 (2004).

[3] K. Nitta *et al.*, JAEA-Data/Code 2010-022 (2010).

[4] T. Misawa *et al.*, *Nucl. Sci. Eng.*, **104**, 53-65 (1991).

採択課題番号 CA24107 固有中性子源を利用した未臨界度測定実験 共同通常
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