CO2-1 Measurement of Energy Resokution in the KURNS-LINAC Pulsed Neutron Facility[2]

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INTRODUCTION: The electron linear accelerator at the Research Reactor Institute, Kyoto University (KURRI-LINAC) had been originally established in 1965 by the High Voltage Engineering Co., USA and started as a 23 MeV machine. In 1971, the machine power had been increased to 46 MeV. The KURRI-linac has two different operation pulse modes. One is a long mode with a maximum repetition rate of 120 Hz, a pulse width of 0.1–4.0 μs and a peak current of about 0.5 A for the measurement at low energies below 10 eV. Another is a short mode with a maximum repetition rate of 300 Hz, a pulse width of 2–100 ns and a peak current of about 5 A for the measurement at high energies above 1 eV. It is worth noting that the peak current of short mode is ten times as large as that of long mode. In measurements of nuclear data, a water-cooled tantalum (Ta) target as a photo-neutron target and a light water moderator are used. There are two kinds of the moderator. One is a water tank type and another is an octagonal shape moderator called "pac-man type". In order to measure accurate nuclear data, it is very important to evaluate the energy resolution (ΔE/E) of a moderator. For example, the energy resolution for pac-man type moderator had been calculated about 0.7 % between energy range of 0.1eV from 10keV [1]. However, measurement and detail evaluation of energy resolutions for the tank type moderator has not carried out in KURRI-LINAC. Furthermore, the relationship between energy resolution and beam pulse width is not discussed in Reference [1].

In this study, the energy resolutions of neutron flux from a tank type moderator were obtained using the TOF technic in 4 operational mode with different pulse width.

EXPERIMENTS: In this study, the energy resolution of KURNS-LINAC was obtained by transmitted neutron flux via ^{181}Ta sample. In order to measure the transmitted neutron flux, a cylindrical moderator tanks which have 19 cm diameter and 30cm height was installed at the target room. The tanks were filled with light water.

We used the neutron at "12 m room" which is located on 135 deg. axis from the electron beam line. The flight path of neutron flux between the ^{181}Ta sample which has 0.2 mm thickness and the moderator was 12.65 m as shown in Fig.1. The transmitted neutron flux was measured by TOF technic with 12 BGO detectors. In order to obtain the relationship between neutron beam pulse width and energy resolution, the KURNS-LINAC was operated under 4 different conditions as shown in Table 1. The beam frequency was adjusted to 50 Hz and the pulse width were about 4 μsec – 0.033 μsec (nominal value).

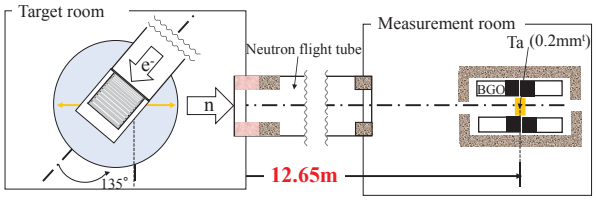

Figure 1 Experimental geometry

Table 1 KURNS-LINAC operational condition in this study.

Run No.	Frequency (Hz)	Beam pulse width (μsec)	Beam current (μA)
1	50	4	20.5
2	50	1	20.5
3	50	0.1	17.5
4	50	0.033	5.6

RESULTS:

Obtained energy resolutions in this experiment are shown in Fig.2 [2]. The energy resolutions with the beam pulse width of 4 μsec are depended on the incoming neutron energy to the sample, clearly. On the other hand, When the pulse width is shorter than 100 ns, the energy resolution shows a constant value. This indicates that as the pulse width increases, the pulse width component is dominant, whereas as the pulse width decreases, the contribution of neutron moderation by the moderator is dominant.

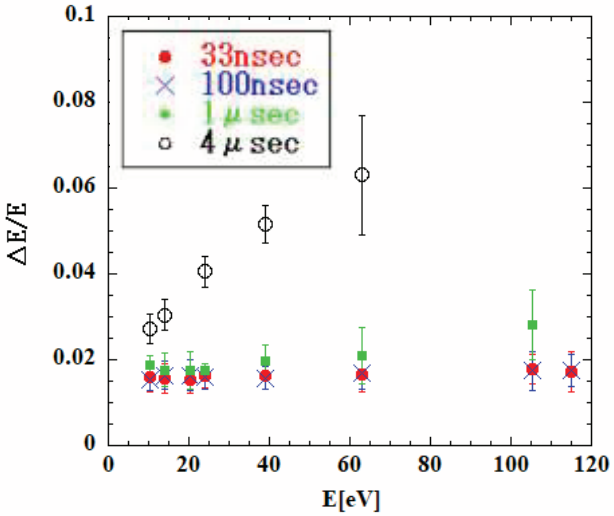

Figure 2 Results of Energy Resolutions

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CO2-2 Measurements of Thermal-Neutron Capture Cross-Section of the ^{237}Np(n, γ) Reaction with TC-Pn in KUR

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INTRODUCTION: In terms of nuclear transmutation studies with neutrons of minor actinides contained in nuclear wastes, accurate neutron capture cross sections are required to obtain transmutation rates. The present work selected ^{237}Np nuclide among minor actinides because ^{237}Np causes long-term radiotoxicity in nuclear waste management due to a long half-life of 2.14×10^6 years.[1] When examining the situation of thermal-neutron capture cross-section (σ_0) of ^{237}Np, there have been discrepancies in reported data even in recent year as shown in **Figure 1**. Previous study [2] has demonstrated that the σ_0 can be measured using the graphite thermal column equipment (TC-Pn) installed in KUR. Consequently, the present work aimed to measure the σ_0 of ^{237}Np by an activation method using a well-thermalized neutron field in the TC-Pn.

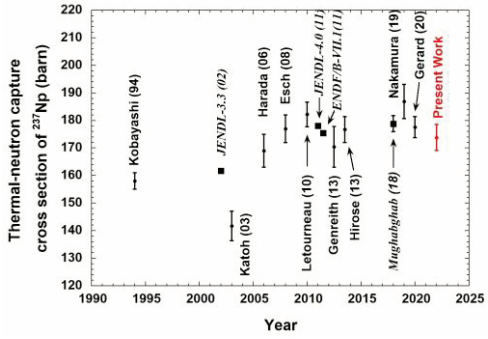

Fig.1 Situation of past reported data for the σ_0 of ^{237}Np.

EXPERIMENTS: Nitric acid solutions of ^{237}Np equivalent to 950 Bq and 370 Bq were dropped onto glass micro filters and dried with an infrared lamp. To monitor neutron flux components, flux monitor sets of Au/Al wire, Co, Sc, Mo and Ta foils were put around each Np sample as shown in **Figure 2**. A set of the ^{237}Np sample and flux monitors was used as a ^{237}Np target; two ^{237}Np targets were prepared. The targets were set separately at the upper edge of polyethylene capsules. Two dummy capsules were transferred to the TC-Pn in advance, and then the capsule containing ^{237}Np target was subsequently transferred so that the target would be exposed to the well-thermalized neutron field in TC-Pn. irradiated for 30 minutes in 1-MW power operation. The similar irradiation was performed on the second ^{237}Np target. After irradiations, the ^{237}Np samples and flux monitors were enclosed in a vinyl bag one by one, and then γ rays emitted from the samples and monitors were measured with a high-purity Ge detector. The sample was placed at a distance of 100 mm from the front surface of the Ge detector. The gamma-ray peak efficiencies of the Ge detector were obtained with a mixed source. In the process of γ-ray measurements, we found that the radiation equilibrium was lost in the ^{237}Np samples. Hence, the amount of ^{238}Np was firstly measured; then the irradiated ^{237}Np samples were stored during a period of 11 half-lives of ^{233}Pa until radiation equilibrium would be maintained again. After that, the ^{237}Np samples were re-quantified with radiation equilibrium.

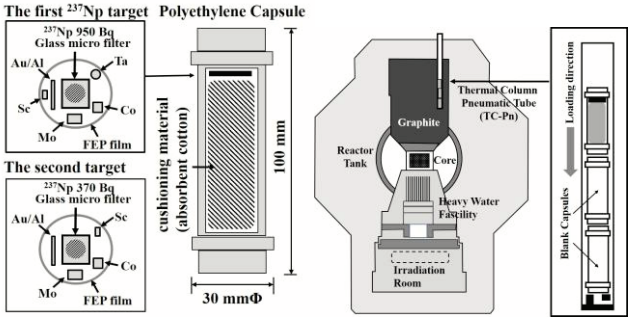

Fig.2 ^{237}Np targets and partial plane structure of KUR.

ANALYSIS: The reaction rates of neutron flux monitors were obtained from their γ-ray yields. With these reaction rates, the thermal-neutron flux component was derived on the basis of *Westcott*'s convention [3], and found to be $(5.69\pm0.09)\times10^{10}$ *n*/cm^2/sec. The ^{237}Np samples were re-quantified using 312-keV γ rays emitted from ^{233}Pa in radiative equilibrium with ^{237}Np. The reaction rate of ^{237}Np was obtained with the sample amount, γ-ray yields given by ^{238}Np and experimental conditions.

RESULTS: The thermal-neutron capture cross section σ_0 was derived straightforward by dividing the reaction rate of ^{237}Np by the thermal-neutron flux component, and found to be 173.7±4.8 barn in the first irradiation and 174.0±5.1 barn in the second irradiation. By taking a weighted average of these results, the σ_0 was finally obtained as 173.8±4.7 barn [4]. The present result was in agreement with the reported data given by a time-of-flight method [5 ,6] within a limit of uncertainty.

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CO2-3 β-decay spectroscopy of rare fission products with a 4π clover detector using an Isotope Separator On-Line KUR-ISOL

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INTRODUCTION: The decay data of the fission products are important for the decay heat evaluation and also structure of neutron-rich nuclei. Many neutron-rich nuclei with mass numbers near 150 do not have detailed decay schemes because of their short half-lives and low fission yields. The nucleus of ^{155}Pr was proposed to have a half-life of 1.49 s by Wu *et al*. [1] and a level scheme of the daughter nucleus ^{155}Nd was reported by Hwang *et al*. using spontaneous fission of ^{252}Cf [2], but, no decay γ-rays have been reported. Based on the experimental results of ^{153,154}Pr in the last year and the expected yields of an on-line mass separator KUR-ISOL, the observation of the γ-rays in the β decay of ^{155}Pr are expected. In this year, to identify γ rays in the β decay of ^{155}Pr, β-γ coincidence measurements were carried out using a high-efficiency clover detector coupled with β-ray detectors. The clover detector has four large Ge crystals with a diameter of 80 mm and a length of 90 mm arranged in the shape of a four-leaf clover around a through hole with a diameter of 15 mm. The β-ray detectors to set in the through hole of the clover detector developed by Ishikawa [3] were modified.

EXPERIMENTS: 72 mg of 93% enriched $^{235}UF_4$ target was inserted at the through-hole facility in Kyoto University Reactor. The nuclei of interest were produced by thermal neutron-induced fission of ^{235}U. The nuclei of interests were transported by He-N_2 gas jets and ionized in a thermal ionization ion source. The mass-separated radioactive beams were collected on a thin Mylar tape and periodically transported to the center of the detector by a computer-controlled tape transport system, and were measured by the detectors. Two identical β-ray detectors were made of plastic scintillators 105 mm long, 12.6 mm wide, and 1 mm thick, contacted with a semi-cylindrical light guide with a radius of 6.5 mm, and set in a through-hole in the clover detector. A 3×3 mm^2 MPPC (Multi-Pixel Photon Counter) module C13367 made by Hamamatsu Photonics was mounted on the end of each light guide. The whole of the detectors was shielded with 10 cm thick lead bricks and 10 cm thick boron-doped polyethylene bricks outside them to reduce background neutrons and γ rays. Data were recorded on APV8008 and APV8016 DSP data acquisition systems made by Techno AP Corporation. The nucleus ^{155}Pr was measured in two modes of collection-measurement time sequence, those are 2.5 s - 2.5 s and 5 s - 5 s, for 16.5 h and 11 h, respectively. Prior to the measurements, ^{94}Rb, ^{94}Sr, ^{94}Y, and ^{146}La were measured to evaluate the performance of the β-ray detectors.

RESULTS and DISCUSSION: The coincidence efficiency of the β-ray detectors was determined from the ratio of the γ-ray peak areas of the coincidence spectra to those of single spectra about the four nuclides. The efficiency of the detector was about over 20% for the energy of β ray above 2 MeV.

To identify the γ rays in the decay of ^{155}Pr, the γ rays and Nd KX-rays and their decay properties, and coincidence relations between KX rays and γ rays were analyzed. The coincidence time was set to 500 ns. The measured β-γ coincidence spectrum is shown in Fig.1. In the spectrum, the background radiation such as the γ ray in the decay of ^{41}Ar produce with ^{40}Ar(n, γ) reaction in the reactor room were reduced effectively. The γ rays in the decay of ^{155}Nd and ^{155}Pm were clearly observed in the spectrum. The seven γ-rays and an excited level were newly assigned in the decay scheme of ^{155}Nd, but unfortunately, any γ rays in the decay of ^{155}Pr were not observed. If γ rays have high multipolarity, the Nd KX-rays should be observed and they should decay with the proposed half-life [2]. So, the decay properties of KX-ray region were analyzed with the spectrum-multi-scaling analysis, but the corresponding energy region of Nd KX-ray did not decay with half-life of about 1.5 s.

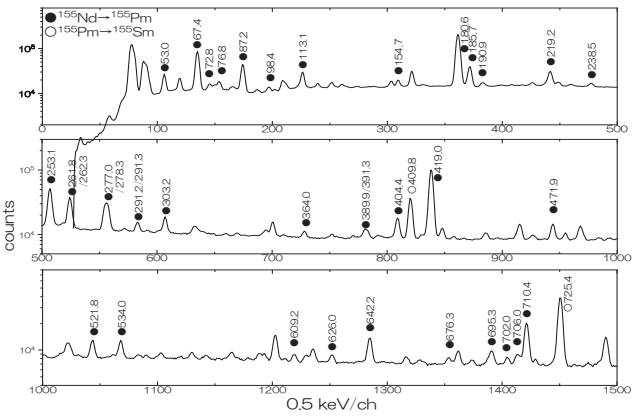

Fig. 1. A β-γ coincidence spectrum for the mass-separated beam of A=155.

The reason why ^{155}Pr could not be observed is the shorter half-life than the reported value [1], the low fission yield than the reference [4], or both. To identify the γ rays, it is necessary to rearrange the measuring setup including the tape transport system.

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CO2-4 Development of a current-mode ^{3}He gas neutron detectors for BNCT

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INTRODUCTION: In the present study, we have developed a current-mode neutron detector [1] that can be used in intense neutron fields such as boron neutron capture therapy (BNCT) fields. It is necessary to measure the neutron flux for a large dynamic range because of connection between a BNCT field in a hospital and a neutron calibration field in the National Institute of Advanced Industrial Science and Technology (AIST). The difference of neutron fluxes between the BNCT field and the calibration field is more than 5 orders of magnitude. We have developed a new ^{3}He gas detector (proportional counter or ionization chamber) with the current mode. The current mode gas detector is expected to be high radiation resistance in comparison with a photo-multiplier tube in the scintillation detector. However, subtraction of gamma-ray component is a problem that need to be overcome in the current-mode neutron detector. In the present study, we experimentally verify the gamma-ray subtraction method in the current-mode ^{3}He gas detector using ^{3}He proportional counters with 4 different gas pressures.

EXPERIMENTS: A collimated neutron beam with 30-mm diameter was obtained by the photo neutron reaction using a tantalum target with a water moderator at the KURNS Linac [2]. Figure 1 shows the experimental setup. In the experiments, we used ^{3}He proportional counters with nominal ^{3}He gas pressures of 4, 0.2, 0.02 and 0.002 atm (Reuter Stokes RS-P4-0806-275, RS-P5-0806-221, RS-P5-0806-220 and RS-P5-0806-222). All counters have total gas pressure with 6 atm and partial pressure of Kr with 2 atm. The ^{4}He gas was used as other gas in the counters with 0.2, 0.02 and 0.002 atm. Each ^{3}He proportional counter was set at approximately 12 m away from the target. The ^{3}He proportional counters were measured by means of the time-of-flight (TOF) method. The measurements were performed for both pulse and current modes. In the pulse mode, signals from the ^{3}He proportional counter were obtained using a pre-amplifier (ORTEC 142PC) and main amplifier (ORTEC 570). In the current mode, the signals were obtained using a current integrator (ORTEC 439). Finally, TOF data were extracted using a multi-stop time to digital converter and a multi-channel analyzer (Fast Com Tec MPA3).

RESULTS: Figure 2 shows results of output count rates for pulse and current modes. The count rates for the linac experiments were normalized using counts obtained from a BF_3 proportional counter used as a neutron monitor. In the case of current mode, the count rate is proportional to the electric current obtained from the counter. In the pulse mode, signals due to only neutrons are obtained by setting regions of interest in the pulse height and TOF spectra. On the other hand, gamma-ray component is included in the current mode. Especially, the gamma flash from the target is dominant. Gamma-ray component is extracted from different of slopes for the pulse and current modes in the linac experiments in Figure 2. In Figure 2, results obtained from measurements at a thermal neutron field at AIST are also included. In this case, the slope of results of pulse mode in the linac experiments are similar to those of current mode in the AIST experiments, because the gamma-ray component in the AIST thermal neutron field is considerably smaller than the gamma flash. It was confirmed that the neutron-gamma discrimination in the current-mode ^{3}He gas detector is available using a pair of ^{3}He and ^{4}He gas detector. To evaluate measurement uncertainty, we will improve further the experimental evaluation of electrical noise and gamma-ray background. Especially, the evaluation of electrical noise is very important in accelerator facilities.

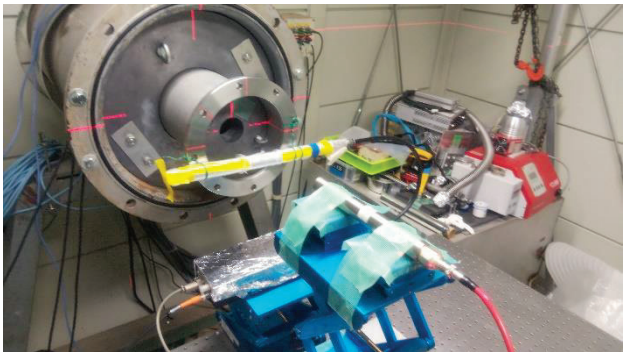

Fig. 1. Experimental setup for the ^{3}He gas detector. The detector was set at approximately 12 m away from the target.

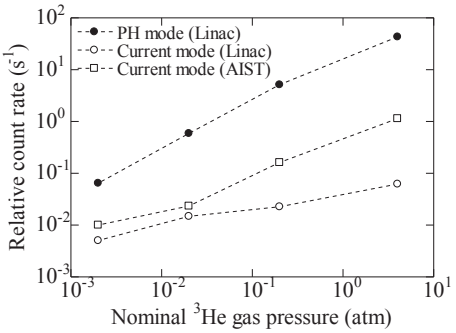

Fig. 2. Outputs from 4 ^{3}He proportional counters in the pulse and current modes.

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CO2-5 Quantitation of Gamma Ray Emission from Caputre Reaction of Uranium-238 (3)

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INTRODUCTION:

In a reactor core, γ rays are radiated by some kinds of neutron induced reactions such as the fission, capture, inelastic scattering, etc. Accordingly, γ ray spectroscopy is potentially useful to identify and quantify those reactions. Focusing on that, one of the authors has studied γ ray spectroscopy for Kyoto University Critical Assembly (KUCA) [1]. In KUCA, the uranium fuel of 93 wt% -^{235}U enrichment is scheduled to be alternated to the fuel with lower enrichment (LEU). By the reduction of the enrichment, the reactions of ^{238}U would be more significant in the core. In order to quantify the capture reaction rates of ^{238}U, we need neutron induced γ ray emission data. For the reason, prompt γ ray spectrum measurements have been conducted for uranium metallic samples irradiated by neutrons of white spectrum at the LINAC facility [1,2].

EXPERIMENTS:

The outline of the γ ray spectroscopy is as same as the previous studies [1,2], where the ^{238}U(n,γ)γ ray was measured with a HP-Ge detector and the pulse height (PH) and the time after the beam pulse incident on the tungsten target (TOF) were stored event by event. Then we can see the PH spectrum for a specific neutron energy. In 2020, PH had been shifted for resonance energy neutrons by extending the pulse width from 1 to 3 μs [2]. In this year, the relation between the pulse width and the spectrum gain was carefully checked. By inserting lead sheet with 6 mm thickness in the neutron beam duct, the spectrum shift was prevented as shown in Fig. 1. Thanks to the extension of the pulse width and the increment of currents, the statistical error in the spectra was reduced.

In order to calibrate the TOF data, we measured TOF for a ^{10}B sample while the source neutron was filtered with Cd, In, Ag, and Co plates. For the uranium sample, PH and TOF spectra were also measured with and without the plates. Those spectra are shown in Fig. 2. The filtered spectrum for uranium sample was often used to evaluate background events. In the TOF range greater than 8000, there is a difference between the spectra of filtered and non-filtered conditions. Those might be due to emission of γ rays from fission products (FPs) with half-lives longer than beam periods (20 ms). FPs are generated by fission of ^{235}U or ^{238}U in the sample. Then we should consider subtracting component from the spectra in Fig.1. Some bumps in spectra for thermal neutrons in Fig. 1 are due to such FP γ rays.

For quantification of γ ray emission per capture, its reaction rate should be known. For that, the pulse height weighting method is useful with calculated response function R. Compared to HP-Ge, response calculation is easier for scintillators. Then we measured ^{238}U(n,γ)γ rays with a BGO of 7.56 cm ϕ x 7.56 cm. The spectra are shown in Fig. 3. For all incident neutron energies, structures related to ^{238}U(n,γ) 4060 keV γ rays appear as well as ^{56}Fe(n,γ) components. We shall estimate the response and try to deduce the reaction rates from these data.

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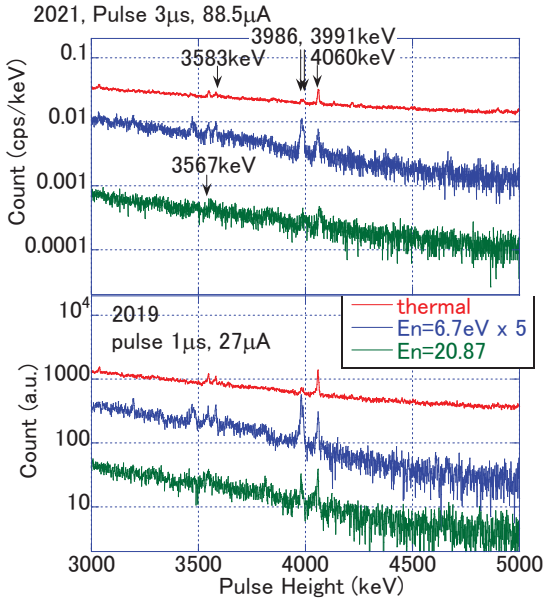

Fig. 1. Comparison of pulse height spectrum for different pulse width.

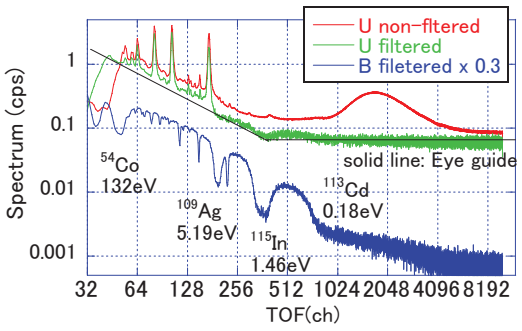

Fig. 2 Filtered TOF spectrum for Background Estimation.

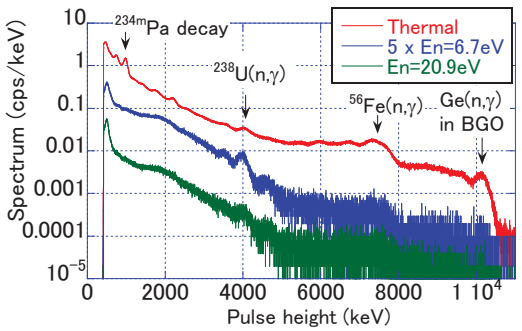

Fig. 3 gamma ray spectrum measured with BGO.

CO2-6 Development of a neutron scintillator for a compact NRTA system (Part 2)

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INTRODUCTION: The Japan Atomic Energy Agency (JAEA) is conducting research and development of active neutron NDA techniques [1] for nuclear nonproliferation and security. As a part of these projects, technology developments for a compact neutron resonance transmission analysis (NRTA) [2] system using a laser-driven neutron source (LDNS) was performed [3]. Using an extremely short pulse laser (~ps), an LDNS can provide short-pulsed fast neutrons (~ns) [4]. This is potentially useful to perform high resolution NRTA measurements (neutron time-of-flight (TOF) measurements) with a short flight path.

In experiments of NRTA, one of the origins of background is the emission of 2.2-MeV gamma rays of $^{1}H(n,\gamma)^{2}H$ reaction in a moderator for pulse neutron generation. The 2.2-MeV gamma-ray emission decreases exponentially just after a neutron pulse generation with a decay time of few hundreds μs. In a measurement using short flight path, therefore, the background overlaps neutron resonance dips of a spectrum, resulting in decline of sensitivity of the NRTA system. To overcome the problem, we developed a multi-layer ^{6}Li glass scintillation (MLS) detector [5] that has low sensitivity to high energy gamma rays.

In the previous experiment [5], we compared the performance of an MLS detector (total ^{6}Li thickness: 0.5 cm) and a conventional ^{6}Li glass scintillation detector (^{6}Li thickness: 1.0 cm). Influence of the 2.2-MeV gamma ray in TOF spectra (<100 μs) were examined. From the study, we concluded that the MLS detector could reduce the sensitivity to 2.2 MeV gamma rays.

Based on the previous results [5], we manufactured a new and upgraded MLS detector. The total thickness of ^{6}Li glass was increased to 1.5 cm to achieve better neutron detection efficiency. Another photomultiplier tube (PMT) was introduced to eliminate noise signals by using the coincidence technique. The performance of the new MLS detector was tested at the Kyoto University Institute for Integrated Radiation and Nuclear Science – Linear Accelerator (KURNS-LINAC). Neutron transmission spectra were measured varying the thicknesses of resonance samples.

EXPERIMENT AND RESULT: A performance test of the MLS detector was performed at the 12 m TOF measurement room of the KURNS-LINAC. Pulsed electron beams of about 30 MeV were used for the pulsed fast neutron generation with a repetition rate of 50 Hz, pulse width of 100 ns, and average current of about 18 μA. To slow down the fast neutrons, a water tank (20 cm in diameter and 30 cm in height) was used as a neutron moderator. A Pb-shadow bar (5 cm in diameter and 20 cm in length) was placed in front of the entrance of the neutron flight tube to reduce the gamma-flash. As the resonance sample, the three indium (In) samples of different thicknesses (0.02, 0.05, and 0.1 mm) were used. To estimate the background level, 0.7-mm thick silver, 0.5-mm thick tantalum, and 1.0-mm thick cadmium were used as the black resonance filters. In the present experiment, the output signals from the MLS detector were processed using a 0.5-GS/s flash waveform digitizer (CAEN V1730) and then transferred to a computer.

Fig. 1 shows the TOF spectra of the three In samples with the resonance filters. The three 1.46 eV resonance peaks of ^{115}In were clearly observed from the TOF spectra obtained by the MLS detector. At present, we have been analyzing the obtained TOF spectra for NRTA.

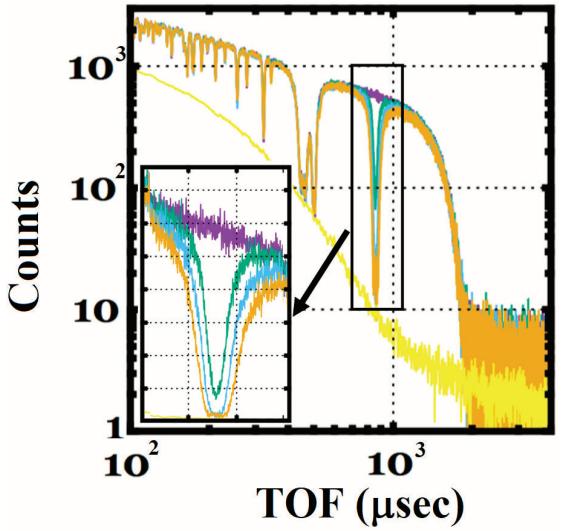

Fig. 1 TOF spectra obtained by the MLS detector.

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CO2-7 Neutron induced fission cross section measurement of ^{233}U

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INTRODUCTION:

The thorium based nuclear fuel cycle has many advantages with respect to the reduction of the risk of proliferation of fissile material and the reduction of buildup of long-lived higher actinides. In the previous works, we have already finished the neutron capture cross section measurement of ^{232}Th [1, 2]. Uranium-233 is the main fuel in the thorium fuel cycle, and its fission reaction plays an important role in the reactor system. Therefore, it is necessary to improve the accuracy of nuclear data associated with the neutron induced fission cross sections of ^{233}U for the feasibility study of the thorium nuclear system. We carried out the neutron induced fission cross-section measurement of ^{233}U.

EXPERIMENTS:

The experiment was performed with the time-of-flight (TOF) method at the 46-MeV electron linear accelerator in Institute for Integral Radiation and Nuclear Science, Kyoto University. Pulsed neutrons were produced from a water-cooled Ta target in a water tank as a neutron moderator. The flight path used in the experiment is in the direction of 135 degrees to the electron beam. The pulsed neutron beam was collimated to 30 mm in diameter. The ^{233}U sample was electrodeposited on a stainless steel plate with an active spot of 20 mm in diameter. The sample deposits of ^{233}U was set in a back-to back (BTB) type double fission chamber [3] which was used as a fission products detector. The distance between the electrode and the deposit layer is 8 mm. The BTB chamber is filled with a mixed gas of Ar and CO_2 at a pressure of 1 atm.

The ^{233}U sample in the BTB chamber was placed at 10.2 m from the Ta target in the TOF beam line. Fission events were detected via the energy deposited in the gas by the fission fragments produced in very thin layer of fissile material. Output signals from the chamber were stored in the Yokogawa's WE7562 multi-channel analyzer as the two dimensional data of pulse-height (PH) and TOF.

The linac was operated with two different conditions. One is for the low-energy experiment below 1 eV with a repetition rate of 50 Hz and the other is for the high energy experiment with a repetition rate of 200 Hz, respectively. In the high energy experiment, a Cd sheet of 0.5 mm in thickness was inserted into the TOF beam line to suppress overlap of thermal neutrons from the previous pulses due to the high frequency of the linac operation. A beam width and an electron energy were 100 ns and about 30 MeV., respectively.

To obtain the neutron flux at the sample position, neutrons were also measured by using a ^{6}Li-glass detector directly. The efficiencies of the ^{6}Li-glass detector were provisionally estimated by using the ^{6}Li(n,α) standard cross sections and the nominal thickness of scintillator. The energy calibration of the TOF neutron beam and the determination of background were performed in the additional experiment with the well-known resonance energies of 1.457 eV of In, 5.19 eV of Ag, 132 eV of Co, and 336 eV of Mn.

RESULTS:

Making use of the BTB chamber and the TOF method, the neutron-induced fission cross sections of ^{233}U were obtained relative to the ^{6}Li(n,α) standard cross section at the energies from 0.002 eV to 1 keV. The relative cross sections were normalized to the reference value of 531.5 b at 0.0253 eV in the JENDL-4.0[4]. Figure 1 shows the comparison of the preliminary results with the evaluated values in JENDL-4.0.

The evaluated data in JENDL-4.0 show general agreement with the present results in the resonance region below 10 eV, although the evaluated data are considerably higher than the present results in high energy region.

We will try to derive the final results from the detail analysis on the detection efficiencies of ^{6}Li-glass detector.

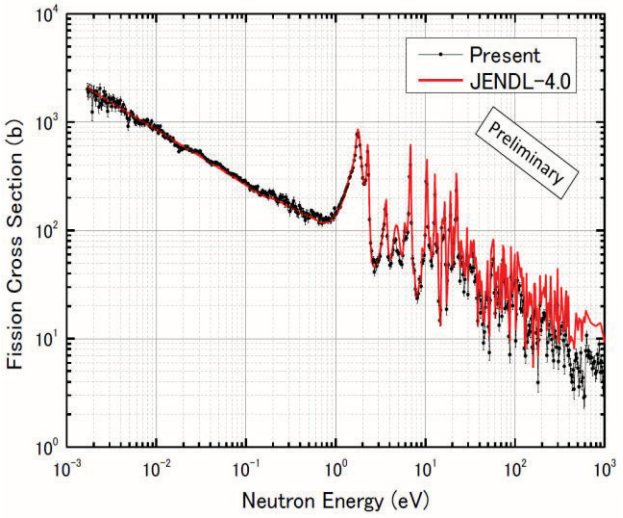

Fig.1 Comparison of the experimental fission cross sections of ^{233}U with the evaluated values in JENDL-4.0.

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