I-1. PROJECT RESEARCHES

Project 7

PR7 Establishment of Integrated System for Dose Estimation in BNCT

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BACKGROUNDS AND PURPOSES:

Several types of accelerator-based irradiation system for boron neutron capture therapy (BNCT) are under development at present. But, there are a number of subjects, which should be improved for the further advance and generalization of BNCT.

In the viewpoints of medical physics and engineering, the advance for dose estimation is one of the important subjects. For the characterization of irradiation field, quality assurance and quality control (QA/QC), clinical irradiation to actual patient, and so on, an ultimate goal is to perform the three-dimensional and real-time dose estimation in discriminating for thermal, epi-thermal and fast neutron doses, gamma-ray dose, and boron dose, with simplicity and low effort. Considering about this ultimate dose estimation, several kinds of dose estimation method are studied. It is so difficult to realize the ultimate dose estimation using only one method, but it is necessary to combine more than two methods.

The purposes of this project research are the advance for the respective dose estimation methods, and the establishment of an integrated system for dose estimation in BNCT.

In the second year of this research project, 2018, the advancement for the respective dose estimation methods were forwarded mainly using Heavy Water Neutron Irradiation Facility and E-3 Neutron Guide Tube at KUR.

RESEARCH SUBJECTS:

The collaboration and allotted research subjects (ARS) were organized as follows;

ARS-1 (30P7-1): Establishment of characterization estimation method in BNCT irradiation field using Bonner sphere and ionization chamber (II). (Y. Sakurai, S. Shiraishi, R. Uchida, T. Takata, H. Tanaka, T. Kawamura, N. Ko, K. Okazaki, M. Sato, A. Sasaki, Y. Kumagai, K. Akita and M. Suzuki)

ARS-2 (30P7-2): Study on new type of neutron spectrometer for BNCT. (A. Ishikawa, A. Uritani, S. Li, K. Watanabe, S. Yoshihashi, A. Yamazaki T. Takada, H. Tanaka and Y. Sakurai)

ARS-3 (29P7-3): Investigation of deterioration characteristics of SOF detector probe. (M. Ishikawa, K. Baba and N. Shimizu)

ARS-4 (30P7-4): Beam profile measurement at E-3 irradiation port by using the self-activation of CsI plate. (A. Nohtomi, R. Kurihara, G. Wakabayashi, Y. Sakurai and T. Takata)

ARS-5 (30P7-5): Characterization of active neutron detector for boron neutron capture therapy. (M. Takada, S. Endo, H. Tanaka, T. Matsumoto, A. Masuda, T. Ueda, M. Ohyauchi, T. Nunomiya, K. Aoyama and T. Nakamura)

ARS-6 (30P7-6): Study for microdosimetry using silicon-on-insulator microdosimeter in the BNCT irradiation field (II). (Y. Sakurai, N. Ko, R. Uchida, T. Takata, H. Tanaka, T. L. Tran, J. Davis, S. Guatelli, A. Rozenfeld, N. Kondo and M. Suzuki)

ARS-7 (30P7-7): Measurement of BNCT beam component fluence with imaging plate. K. Tanaka, Y. Sakurai, T. Kajimoto, Y. Murakami, Y. Ito, H. Tanaka, T. Takata and S. Endo)

ARS-8 (30P7-8): Development of neutron fluence distribution measuring device using thermoluminescence slabs. (K. Shinsho, S. Yanagisawa, R. Oh, G. Wakabayashi Y. Koba and H. Tanaka)

ARS-9 (30P7-9): The study for development and application of tissue equivalent neutron dosimeter. (M. Oita, T. Kamomae, N. Hayashi, T. Takada and Y. Sakurai)

ARS-10 (30P7-10): Development and evaluation of 3D polymer gel dosimeter for the measurement of dose distribution in BNCT. (S. Hayashi, Y. Sakurai, M. Suzuki, T. Takata and R. Uchida)

ARS-11 (30P7-11): Establishment of beam-quality estimation method in BNCT irradiation field using dual phantom technique (II). (Y. Sakurai, T. Takata, H. Tanaka, N. Kondo and M. Suzuki)

ARS-12 (30P7-12): Development of a prompt gamma-ray imaging detector for boron neutron capture therapy. (K. Okazaki, K. Akabori, T. Takata, S. Kawabata, Y. Sakurai and H. Tanaka)

ARS-13 (30P7-13): Radiation damage experiment on novel scintillator material and study on material for development of irradiation monitor in BNCT. (S. Kurosawa, A. Yamaji, T. Horiai, S. Kodama, S. Yamato and H. Tanaka)

ARS-14 (30P7-14): Establishment of the Imaging Technology of 478 keV Prompt Gamma-rays of Boron-neutron Capture Reaction and the Measurement of the Intensity of the Neutron Field. (M. Ishimoto, T. Mizumoto, S. Sonoda, A. Takada, T. Tanimori, T. Takata and Y. Sakurai)

ARS-15 (30P7-15): Feasibility study for establishing quality assurance and quality control for radiation field in boron neutron capture therapy. (S. Nakamura, A. Urushiyama, T. Nishio, M. Tsuneda, T. Masuda, A. Sano, H. Okamoto, S. Nishioka, K. Iijima, M. Takemori, H. Nakayama, Y. Sakurai, H. Tanaka, T. Takata, M. Suzuki and J. Itami)

ARS-16 (30P7-16): Patient-position monitoring by using Kinect sensor for boron neutron capture therapy. (T. Takata, H. Tanaka, Y. Sakurai, Y. Tamari and M. Suzuki)

For ARS-10, no results were obtained because no machine time were used due to the schedule of the collaborators. So, the report of this research subject is not appeared.

PR7-1 Establishment of characterization estimation method in BNCT irradiation field using Bonner sphere and ionization chamber (II)

Y. Sakurai, S. Shirai-shi[1], R. Uchida[1], T. Takata, H. Tanaka, T. Kawamura[1], N. Ko[1], K. Okazaki[1], M. Sato[1], A. Sasaki[1], Y. Kumagai[1], K. Akita[1,2], M. Suzuki

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INTRODUCTION: Research and development into several types of accelerator-based irradiation systems for boron neutron capture therapy (BNCT) is underway [1,2]. In the near future, BNCT using these newly developed irradiation systems may be carried out at multiple facilities across the world. Considering this situation, it is important that the estimations for dose quantity and quality are performed consistently among several irradiation fields, and that the equivalency of BNCT is guaranteed, within and across BNCT systems. Then, we are establishing the quality assurance and quality control (QA/QC) system for BNCT irradiation field.

As part of the QA/QC system, we are developing estimation method for neutron energy spectrum using Bonner sphere [3]. For our spectrometer using Bonner sphere, liquid such as pure water and/or boric acid solution is used as the moderator. A multi-layer concentric-sphere case with several sphere shells is prepared. The moderator and its diameter are changeable without entering the irradiation room, by the remote supply and drainage of liquid moderator in the several layers. For the detector, activation foils are remotely changed, or online measurement is performed using SOF (scintillator with optical fiber) detector containing boron, etc. [4].

In 2018, the detailed optimization was performed mainly by simulation for the development of the Remote-changeable Bonner-sphere Spectrometer (RBS). Concurrently, estimation experiment was performed for the irradiation characteristics at Heavy Water Neutron Irradiation Facility of Kyoto University Reactor (KUR-HWNIF).

METHODS: In the neutron energy spectrometry by Bonner-sphere and activation foils, the combinations of the moderator material and diameter should be previously decided and prepared. Of course, the more information can be obtained as the more moderators and detectors are prepared. However, the information number from those measured data is less than the combination number, because of the overlapped regions among the combinations. The selection is important, in which the more information number is obtained for the combination number.

The combination of moderator and detector is decided, for that the response functions cannot be approximated by the linear functions of the other response functions. The accuracy and precision for the spectrometry can be higher, because the independent information can be obtained from the measurement by the respective combinations. We were developed the selection method, High Independence Selection (HIS) [5].

On the assumption of the application in a typical BNCT irradiation field, the combination of the moderators for boron-10 concentration and diameter was optimized by HIS. Ten types of acrylic container for Bonner sphere is assumed. These container have five hollow shells for supplying moderators and acrylic walls. The thicknesses for the shells and walls in each container are 1 cm and 1 mm, 1 cm and 2 mm, 1 cm and 3 mm, 1 cm and 4 mm, 1 cm and 5 mm, 2 cm and 1 mm, 2 cm and 2 mm, 2 cm and 3 mm, 2 cm and 4 mm and 2 cm and 5 mm. Moderators are pure water and boron acid water (0.14wt%, B-10 solubility at 20 degree centigrade).

Thirty-three combinations for the shells and moderators are simulated. For the ten combinations, only one shell is filled with pure water or boron acid water, and the remaining four shells are empty. For the twenty three combinations, the shells are filled with the moderators in turn from the central shell. Totally, 330 patterns for the spectrometer are simulated.

RESULTS: Some optimized structures of RBS were selected by HIS. Figure 1 shows one of the optimized combinations for RBS.

CONCLUSION: We have a plan to create the RBS, based on the optimization result. Additionally, we have a plan to perform the spectrometry experiments at KUR-HWNIF, etc., in order to confirm the efficacy of RBS.

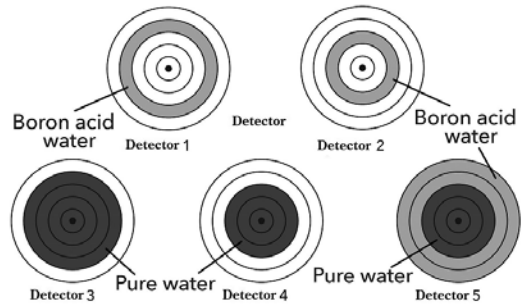

Fig. 1. One of the optimized combinations for the remote-changeable Bonner-sphere spectrometer.

REFERENCES:

[1] H. Tanaka *et al.*, Nucl. Instr. Meth., B **267** (2009) 1970-1977.
[2] H. Kumada *et al.*, Appl. Radiat. Isot., **88** (2014) 211-215.
[3] H. Ueda *et al.*, Appl. Radiat. Isot., **104** (2015) 25-28.
[4] M. Ishikawa *et al.*, Radiat. Oncol., **11** (2016) **105** (1-10).
[5] H. Ueda, Doctoral Thesis (2016).

Study on New Type of Neutron Spectrometer for BNCT

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INTRODUCTION: Boron neutron capture therapy (BNCT) is one of the promising treatment methods for cancers such as brain tumors. In recent years, accelerator-driven neutron sources have been developed because of its simplicity of management. In commissioning of these facilities, the irradiation field should be characterized in order to assure designed specifications, such as neutron intensity, the neutron energy spectrum and gamma-ray contamination.

We are developing a new neutron energy spectrometer using an optical fiber type detector. A conventional Bonner sphere neutron spectrometer has only less than ten detectors, typically five or six detectors. Moderation based neutron spectrometers commonly has relatively low response for epi-thermal neutron energy region. One of the solutions to improve the energy response is to increase the number of detectors with different responses. Tamaki et al. proposed a liquid-moderator-based neutron spectrometer, in which liquid moderator volume is changed to make various responses [1]. On the other hand, we propose the spectrometer consisting of a liquid moderator and a thermal neutron flux profile scanner using an optical fiber type detector [2].

In this study, we experimentally conducted response evaluation of the water-moderator-based neutron spectrometer using a thermal neutron profile scanner the optical fiber type detector. Figure 1 shows the response function of the water-moderator-based neutron spectrometer with 200x200x200 mm^3 size calculated by PHITS code.

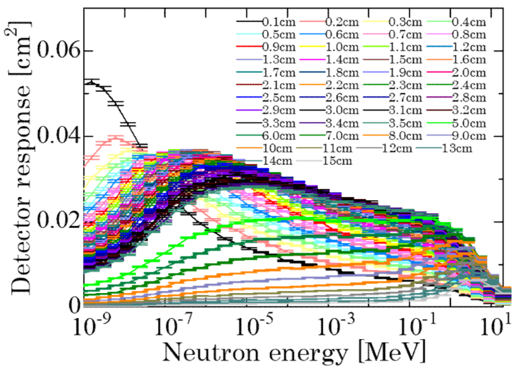

Fig. 1. Response function of the water-moderator-based neutron spectrometer.

EXPERIMENTS: The water phantom with 200 x 200 x 200 mm^3 size was irradiated with epi-thermal neutrons at Heavy Water Irradiation Facility of Kyoto University Reactor. The thermal neutron profile in this phantom was measured by the optical fiber type neutron detector using a small Eu: $LiCaAlF_6$ scintillator. The detector head was moved with a linear stage. The gold foil activation method was also conducted to measure the thermal neutron profile. In order to evaluate pure thermal neutron flux, the Cd subtraction method was used.

RESULTS: Figure 2 shows the depth profile of the thermal neutron flux in a water phantom measured by the gold foil activation method and the optical fiber type detector. Both results well agreed each other. In Fig. 3, we quantitatively compared the both results. The results of the optical fiber detector well reproduce ones of the gold foil activation method within 10% difference.

REFERENCES:

[1] S.Tamaki, *et al.*, Nuclear Instruments and Methods in Physics Research Section A , **870**, 90 (2017)
[2] K. Watanabe *et al.*, Nuclear Instruments and Methods in Physics Research Section A , **802**, 1 (2015)

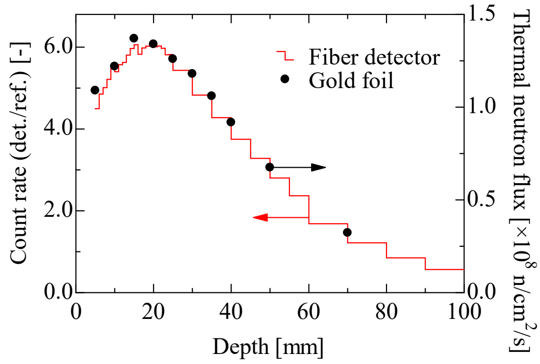

Fig. 2. Depth profile of the thermal neutron flux in a water phantom measured by gold foil activation method and the optical fiber type detector.

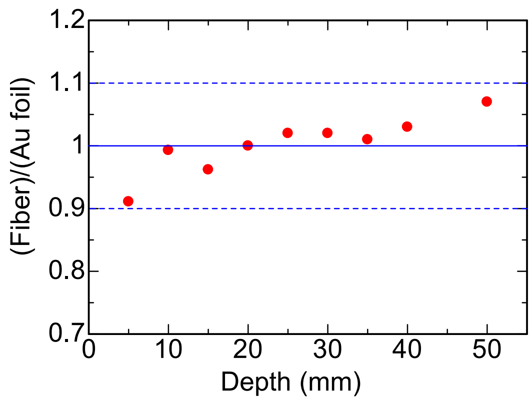

Fig. 3. Comparison between the thermal neutron flux measured by gold foil activation method and the optical fiber type neutron detector.

PR7-3 Investigation of deterioration characteristics of SOF detector probe

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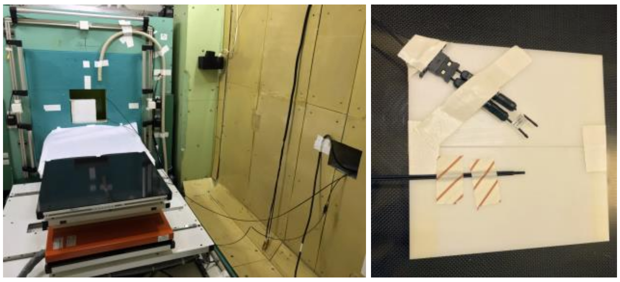
INTRODUCTION: In the conventional BNCT, thermal neutron flux and thermal neutron fluence during treatment could not be measured in real time because gold wire activation was used to evaluate thermal neutron fluence. Therefore, we developed a detector (SOF detector; Scintillator with Optical Fiber Detector) with a plastic scintillator attached to the tip of the optical fiber, tried real-time measurement of thermal neutron flux in neutron capture therapy, and got good results. The long-term stability of the SOF detector and the wide measurement dynamic range (linearity of 10^4 to 10^{10} n/cm^2/s) have been confirmed in previous collaborative experiments. [1,2]
However, signal degradation of SOF detector in long-term exposure was reported [3]. The signal degradation might not be a significant problem in case that calibration can be performed before use. However, signal degradation greatly affects measurement accuracy in case of long-term monitoring because calibration prior to use is difficult. Therefore, in this research, in order to identify the cause of signal degradation, identification of the substance causing the signal degradation was attempted by devising the combination of the materials constituting the probe.

EXPERIMENTS: The experimental geometry is shown in Fig. 1. The Heavy Water Neutron lrradiation Facility (HWNIF) of KUR was used for irradiation. The SOF detector probes were arranged at a surface of a 30 x 30 x 5 cm thick polyethylene block which was placed in the irradiation field, and signal change was monitored online. The combinations of neutron converter and reflectors contained in the irradiated probes are shown in Table 1. All probes used BC490 as a plastic scintillator, and Probes 1 to 3 were mixed with ^{6}LiF as a neutron converter. Moreover, BC642 (PTFE Reflector Tape) and BC620 (Reflector paint) were used as a reflective material for Probe 1 and 2, respectively. Probe 4 used BC620 reflector without any neutron converter. As a reference probe, the probe which consist of BC490 with BC642 was used.

Table 1. Irradiated sample probes with and without neutron converter and reflectors.

Probe #	Neutron converter	Reflector
1	^{6}LiF	BC642
2	^{6}LiF	BC620
3	^{6}LiF	None
4	None	BC620
Reference	None	BC642



(a) Overview of irradiation (b) detector layout

Fig. 1. Irradiation geometry for on-line assessment of deterioration characteristics.

RESULTS: Figure 2 shows the change in signal strength of the SOF detector by on-line monitoring. Each signal is normalized to the average value of the entire measurement. From Fig. 2, the change of signal intensities of Probes 1 to 3 were relatively decreased, whereas the signal intensity of Probe 4 was relatively increased with neutron irradiation. The 1.64% signal decrease of the Probe 1 which is used in conventional SOF detector was observed from the start of irradiation. This suggests that the deterioration of ^{6}LiF used as a neutron converter may cause a decrease in signal intensity. From the result of Probe 4, it was also found that the reflective material of BC620 reflector paint might be activated by irradiation of thermal neutron.

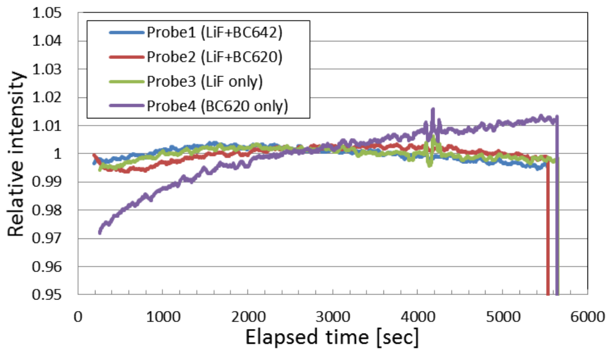


Fig. 2. Irradiation geometry for off-line assessment of deterioration characteristics.

REFERENCES:
[1] M. Ishikawa *et al.*, Appl. Radiat. Isot., **61** (2004) 775-779.
[2] M. Ishikawa *et al.*, Nucl. Instr. Meth., A **551** (2005) 448-457.
[3] M. Komeda *et al.*, Appl. Radiat. Isot., **67** (2009) 254-257.

PR7-4

Beam profile measurement at E-3 irradiation port by using the self-activation of CsI plate

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INTRODUCTION: In our previous studies, neutron intensity distribution on a CsI scintillator plate has been observed by a CCD camera and analyzed by using a "decaying self-activation imaging technique" [1, 2]. The decaying self-activation imaging, which has been proposed recently by our group, is based on the analysis of time variation of specific radio-activities generated and remained inside the CsI plate after the termination of neutron irradiation. In the present work, this technique was applied to measure the thermal neutron beam profile at E-3 irradiation port facility of KUR, which is mainly utilized for prompt gamma-ray analysis to evaluate B-10 concentration in a sample.

EXPERIMENTS: A CsI plate (75mm x 75mm x 2mm) which is packed with aluminum container and glass window was irradiated at E-3 irradiation port facility of KUR. After the termination of 75 min. irradiation at the exit collimator of LiF (opening H:14mm x V:18mm). The luminance distributions of a CsI plate are recorded every one minute as a series of images by a cooling type CCD camera (SBIG:STF8300M) with a telescope lens (SIGMA : 30mm F1.4 DC) in a black box. A gold foil was used to evaluate the fluence rate of thermal neutron by the activation analysis.

RESULTS: As the result of activation analysis of gold foil, thermal neutron fluence rate at the irradiation port was determined to be 1.83 x 10^5[n/cm^2/s]. Figure 1 shows a typical I-128 luminance image observed by the CCD camera. The I-128 luminance component was extracted by the decaying self-activation imaging technique. Yellow rectangular indicates 10% level region of the maximum luminance value; the horizontal length is 15 mm and the vertical length is 21 mm.

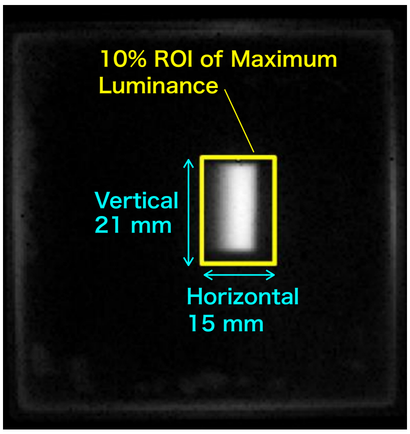

Fig. 1. Typical I-128 self-luminance image observed by a CCD camera.

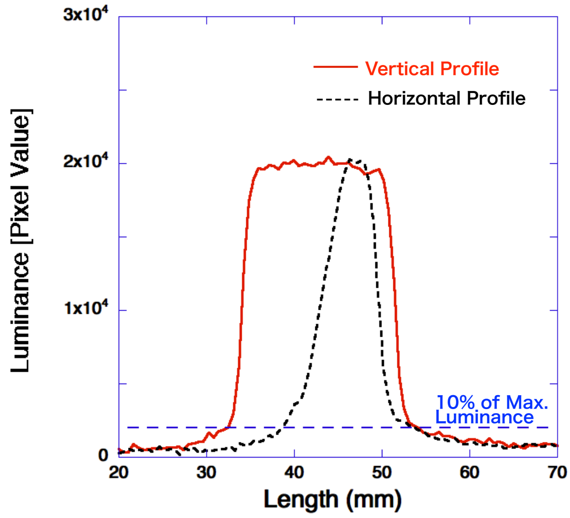

Fig. 2. Horizontal and vertical beam profile of luminance observed by a CCD camera.

So, those are almost consistent with the actual physical dimension of the collimator opening. However, as shown in Fig. 2, horizontal beam profile shows an antisymmetrical peak. Figure 3 displays a 3-D graphical presentation of the beam profile of luminance observed.

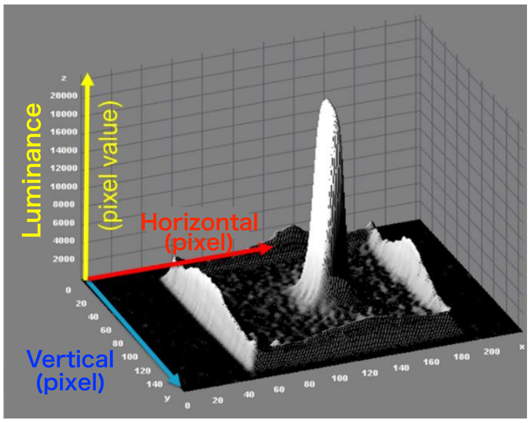

Fig. 3. Three-D graphical presentation of the beam profile observed.

From the results described above, it has been confirmed that the thermal neutron beam profile at E-3 irradiation port facility is almost uniform and symmetrical along the vertical direction. On the other hand, along the horizontal direction, the beam profile indicates a shape like distorted Gaussian distribution and dose not have flat region.

ACKNOWLEDGEMENT: This work was supported by JSPS KAKENHI Grant Number JP16K10320.

REFERENCES:
[1] A.Nohtomi *et al.*, Nucl. Instr. and Meth., **A832** (2016) 21-23.
[2] A.Nohtomi *et al.*, JPS Conf. Proc., **24** (2019) 011041.

PR7-5 Characterization of Active Neutron Detector for Boron Neutron Capture Therapy

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INTRODUCTION: Boron Neutron Capture Therapy, which is called as BNCT, is a binary radiotherapy method developed to treat patients with certain malignant tumors. BNCT has been widely used at nuclear reactors, however, several accelerator-based BNCT facilities have been developed / ready to clinical treatment for more accessible for patients. Measurements of neutron fluences irradiated to the patients are important for BNCT success. The neutron fluence has been measured using passive method of gold neutron activation. In the accelerator-based BNCT, intensity of neutron beam could be varied with time due to instability of the accelerator condition and loss of neu-tron target. We should continuously measure neutron beams in a real time.

For this purpose, we have developed active neutron detector to measure BNCT beam directly, which is named as DAD-BNCT. It can measure both intense neutron beam over 1×10^9 (n cm^{-2} s^{-1}) and high dose-rate gamma rays around 500 mGy/h [1], simultaneously. The detector consists of thin silicon diode having 40 μm in thickness [2] and thin LiF coated on Teflon plate. Intense neutron beam can be measured by detecting triton produced by the $^6Li(n,t)^4He$ reaction at the LiF neutron converter. In this study, we characterized the active neutron detector for BNCT beam.

EXPERIMENTS: The neutron measurements were performed at the heavy water irradiation facility of research reactor of Institute for Integrated Radiation and Nuclear Science, Kyoto University [3]. The DAD-BNCT was located without any neutron and gamma-ray shields in the irradiation room of this facility. Signals fed from the DAD-BNCT were charge amplified with the commercial preamplifier and main amplifier, and then signals were acquired with a multi-channel analyzer. The thermal neutron beam fluxes and the neutron energy spectrum are changed by the aperture area with Cd plate slits at an upward of the irradiation location. The thermal neutrons from 2×10^7 to 1×10^9 (n cm^{-2} s^{-1}) were irradiated to the DAD-BNCT. The response functions are obtained from differences of the measurements under the neutron fields with different areas of Cd slits. Also, the response functions with/without the LiF neutron converter were experimentally obtained to evaluate the responses of detecting secondary particles produced by only the $^6Li(n,t)^4He$ reaction. Bias dependence of the response functions were measured.

RESULTS: The response functions, normalized with the thermal neutron flux, measured under the neutron irradiation with the 600-mm-width Cd slit is shown in **Figure 1**. This setup can irradiate the detectors with highest thermal neutron flux. A clear peak around 650 ch., labeled as (A) in Fig.1, and an edge around 550 ch., labeled as (B), in the spectrum are created by tritons, produced by the $^6Li(n,t)^4He$ reaction, fully stopped in and transmitted through the depletion layer of silicon diode, respectively. Increasing the neutron flux, position of this peak is constant; while, peak widths are slightly broader. Integrating the count rate in this peak region, the detection efficiency of DAD-BNCT to thermal neutron beam was also evaluated. The results are 2.9×10^{-6} (# cm^2), temporarily. The continuum from 200 to 550 ch., labeled as (C), is created by alpha particles from the neutron and boron reaction. Below 200 ch., labeled as (D), these events are created by detecting gamma rays. Therefore, the DAD-BNCT can measure BNCT neutron beam directly, well separated from gamma-ray events. Neutron damage have never been observed through the BNCT experiments.

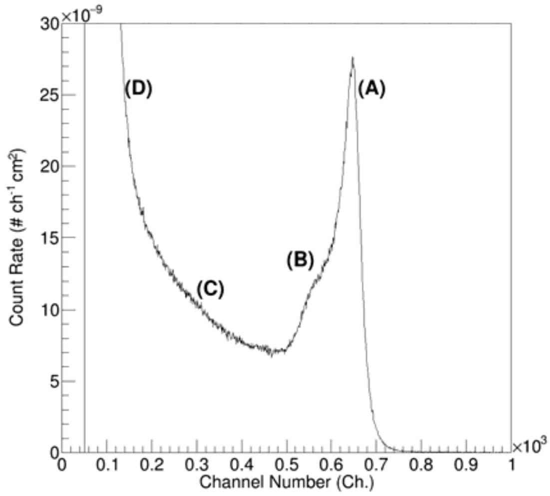

Figure 1. Neutron response functions of DAD-BNCT for BNCT neutron beam under a 600-mm-width Cd slit.

REFERENCES:

[1] M. Takada *et al.*, prepared.

[2] M. Takada *et al.*, Radiat. Meas., 99, 33-54 (2017).

[3] T. Kobayashi *et al.*, Nucl. Technol., 131, 354-378 (2000).

PR7-6 Study for microdosimetry using silicon-on-insulator microdosimeter in the BNCT irradiation field (II)

Y. Sakurai, N. Ko[1], R. Uchida[1], T. Takata, H. Tanaka, T. L. Tran[2], J. Davis2, S. Guatelli[2], A. Rozenfeld[2], N. Kondo, M. Suzuki

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INTRODUCTION: Research and development into several types of accelerator-based irradiation systems for boron neutron capture therapy (BNCT) is underway [1,2]. In the near future, BNCT using these newly developed irradiation systems may be carried out at multiple facilities across the world. In contrast to conventional radiotherapy, the types of radiation present in BNCT consists of many distinct radiation components, each having a different biological weighting factor.
Microdosimetry is an effective dosimetry technique in a mixed radiation environment. Using this technique, it is possible to derive the relative contributions of different radiation modalities. The feasibility study of a novel 3D mesa bridge silicon-on-insulator microdosimeter (SIM) in BNCT [3], developed by University of Wollongong (UOW).
In 2018, the more detailed optimization was performed for SIM by Monte Carlo simulation. And also, the experiments for the confirmation of the characteristics of SIM were performed using Heavy Water Neutron Irradiation Facility (HWNIF) and E3 neutron guide tube (E3) of Kyoto University Reactor (KUR).

METHODS: The bridge microdosimeter is comprised of an array of 4248 individual silicon cells fabricated on a 10 μm thick n-type silicon-on-insulator substrate. Figure 1 shows the outline of SIM.
For the optimization simulation, the different boron converter and silicon-on-insulator substrate thickness was modelled and the energy deposition within the detector was simulated using the Particle and Heavy Ions Transport Code System (PHITS). The T-deposit tally in PHITS was used to calculate the energy deposited per event inside the sensitive volume of the bridge microdosimeter. The lineal energy was calculated by dividing the deposited energy per event by the average chord length of the detector. The clinical BNCT field at KUR-HWNIF using both thermal and epithermal irradiation modes were used in this study.
For the experiment at HWNIF, one of the aims was the confirmation for the response characteristics of SIM at the epi-thermal neutron field with larger irradiation area. The other of the aims was the data acquisition for the depth distributions for the SIM response.
For the experiment at E3, the aim was the confirmation for the response characteristics of SIM at the thermal neutron field with smaller irradiation area.

RESULTS: Figure 2 shows one of the microdosimetric spectrum obtained from the optimization simulation. This simulation results showed that the sensitivity of SIM was increased by the boron converter especially in the thermal neutron irradiation field.
For the both experiments at HWNIF and E3, no valid data was obtained foe some reasons. At HWNIF, the effectiveness of neutrons to the electronics of SIM was so large that the data acquisition was stopped immediately after the acquisition start. At E3, the background noise was so large that the response signal of SIM was buried.

CONCLUSION: The simulation results show that this microdosimeter can be utilized as an effective tool for dosimetry in BNCT field. The experiments are planed using KUR again.

Fig. 1. Outline of silicon-on-insulator microdosimeter.

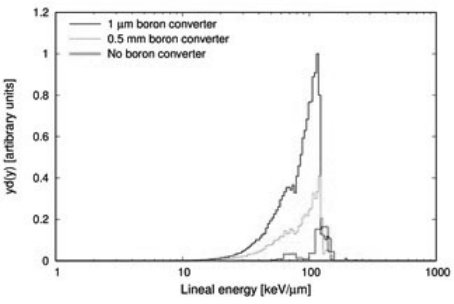

Fig. 2. Microdosimetric spectrum for the boron converter thickness.

REFERENCES:
[1] H. Tanaka *et al*., Nucl. Instr. Meth. B **267** (2009) 1970-1977.
[2] H. Kumada *et al*., Appl. Radiat. Isot., **88** (2014) 211-215.
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PR7-7

Measurement of BNCT beam component fluence with imaging plate

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INTRODUCTION: Dose evaluation is required for quality assurance in the irradiation field for boron neutron capture therapy. This study investigated the use of the imaging plate (IP) combined with beam component converter.

EXPERIMENTS: The converter configuration chosen for this study is shown in Fig. 1. The IP is BAS-TR from Fuji Film corporation, Japan. The IP #1 in carbon is for gamma rays, #2 for epithermal neutrons, and #3 for fast neutrons. Here, material #2 is polyethylene. Material #1 is polyethylene infused with LiF, where ^{6}Li is enriched up to 95 at% The concentration of ^{6}Li in Material #1 is 10 wt%. The details of the converter was described previously [1].
The experiment was performed with the standard epithermal neutron irradiation mode of KUR-HWNIF at 1 MW. The beam size was set to about 120 × 120 mm^2 using the collimator. The irradiation was performed for 2 minutes. The nominal value of the flux at the center of the collimator aperture was 7.07 × 10^6 cm^{-2}s^{-1}, 1.33 × 10^8 cm^{-2}s^{-1} and 1.38 × 10^7 cm^{-2}s^{-1} for thermal, epithermal and fast neutrons, respectively. The gamma ray flux was 1.25 × 10^7 cm^{-2}s^{-1}.

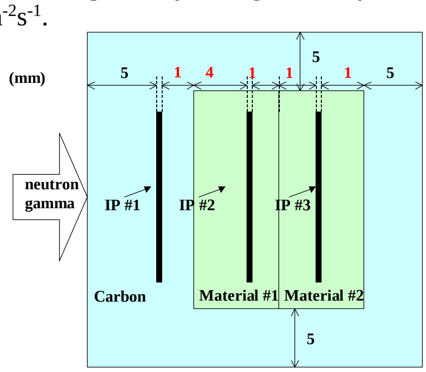

Fig. 1 Converter configuration.
The figure is not in scale.

The fluence ϕ_j of each component was determined using the following model;

$$PSL = \begin{pmatrix} PSL_1 \\ PSL_2 \\ PSL_3 \end{pmatrix} = \begin{pmatrix} a_{11} & a_{12} & a_{13} \\ a_{21} & a_{22} & a_{22} \\ a_{31} & a_{32} & a_{33} \end{pmatrix} \begin{pmatrix} \phi_1 \\ \phi_2 \\ \phi_3 \end{pmatrix} = A \cdot \phi \quad (1),$$

$$\phi = A^{-1} \cdot PSL \quad (2).$$

where a_{ij} denotes the sensitivity of the ith IP for the component j.

RESULTS: The fluence estimated using the results for the three IPs are shown in Fig. 2. All of three beam components yielded positive values successfully. This was intended by setting the each beam component contribution to the total of the energy deposition to the IP at about 10 % or more. The estimated fluence showed reasonable distributions where the values around the beam center was higher than others. The reproducibility of the result is planned to be investigated.

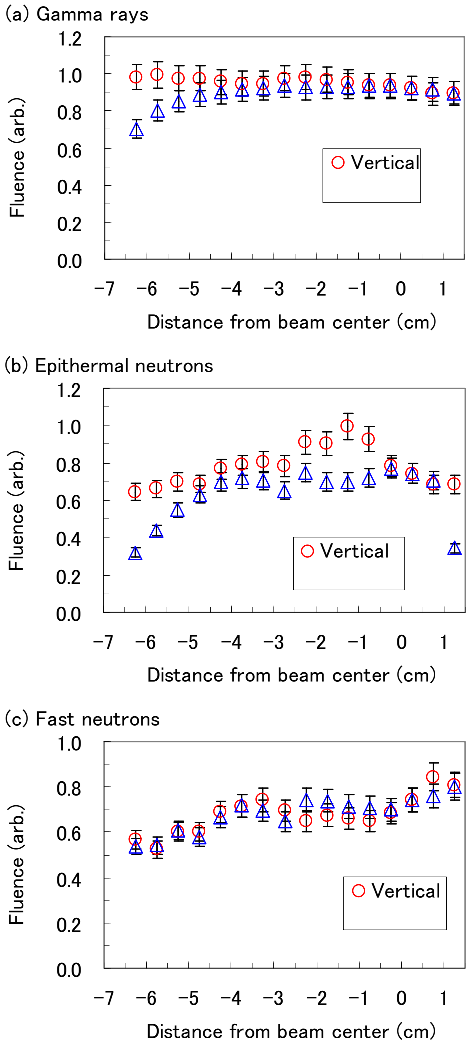

Fig. 2 Estimated fluence distribution

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PR7-8

Development of Neutron Fluence Distribution Measuring Device using Thermoluminescence Slabs

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INTRODUCTION: Boron Neutron Capture Therapy (BNCT) is one of the radiation therapies using neutrons and ^{10}B drugs which are attracted to tumors. BNCT is expected to be next-generation cancer therapy which will improve the QOL of patient because it is able to irradiate a cancer cell at the molecular level selectively. However, dosimetry techniques in mixed neutron-gamma fields have not been established yet. Therefore we examine the possibility that two dimensional Cr doped Al_2O_3 thermoluminescent slabs can apply to dosimetry in mixed neutron-gamma field. The Cr doped Al_2O_3 thermoluminescent slabs dosimeter have good properties for photons. [1,2] For example, it has a high sensitivity and high special resolution. In this study, we were prepared Cr doped Al_2O_3 thermoluminescent slabs of 10 × 10 × 0.7 mm^3 and 80 × 80 × 0.7 mm^3, then they were investigated TL properties for mixed neutron-gamma field.

EXPERIMENTS: Low melting point Al_2O_3 of Chibaceramic MFG Co. LTD., which was composed of Al_2O_3 > 99.5 wt%, SiO_2 < 0.10 wt%, Fe_2O_3 < 0.05 wt%, Na2O < 0.10 wt%, Cr < 2ppm, Cd < 1ppm, Pb < 1ppm, Hg < 1ppm was used. The bulk density of the plates was $3.7 g\cdot cm^{-1}$. The dimensions used for the glow curve measurements were 10 × 10 × 0.7 mm^3. The concentration of Cr_2O_3 in the present study was 0.05 wt%. The assumed irradiation fields are the standard thermal neutron irradiation mode, mixed neutron irradiation mode and epithermal neutron irradiation mode in KUR-HWNIF, with a power of 1MW. The glow curves were recorded from room temperature up to 400 °C at a heating rate of $0.1\ °C\cdot s^{-1}$. The Two-dimensional TL measurement system was made by Seisei Manufactory Co.,Ltd. It consists of a complementary metal oxide semiconductor (CMOS) camera (ORCA®-Flash4.0 V2, C11440 -22CU) , 80 × 80 mm^2 heater (Sakaguchi E.H VOC CORP.), and a dark box. After exposure (irradiation field size of 100 × 100 mm^2), the TL slabs were heated to 400 °C for 5 min. The TL images were captured using a CMOS camera equipped with a thermal cut filter.

RESULTS: Figure shows the arrangement of TL slabs, TL phosphor BeOs and ^{197}Au foils (upper) , and the TL imaging (lower). Table shows the thermal neutron fluencies and γ- ray dose for each numbers in Fig.. From these results, we were clear that Cr doped Al_2O_3 thermoluminescent slab has sensitivities for thermal neutron and γ-ray. The TL imaging was high special resolution. Not only thermal neutron and γ- ray imaging but also the β-ray imaging from the $^{197}Au(n,\gamma)^{198}Au \rightarrow {}^{198}Hg$ was able to acquire it clearly.

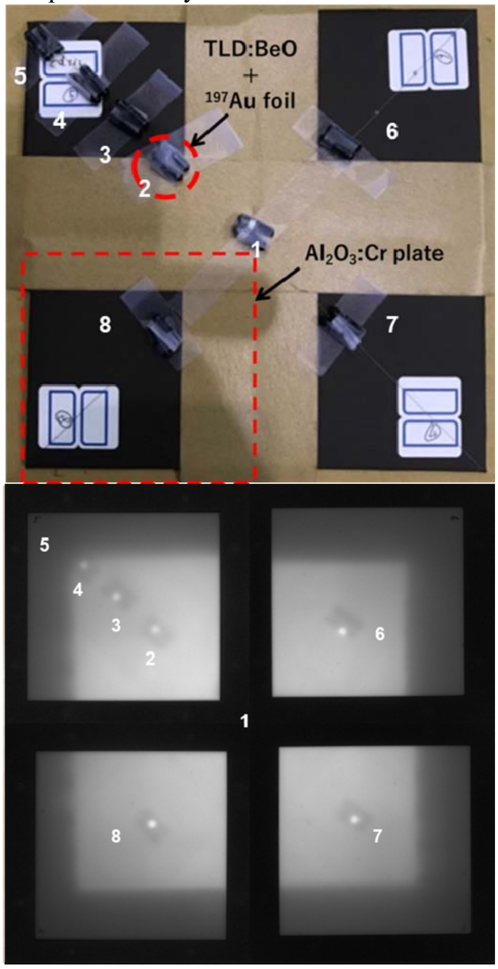

Fig. Arrangement of TL slabs, TL phosphor BeOs and ^{197}Au foils (upper) , and the TL imaging. (lower).

Table Thermal neutron fluencies and γ- ray dose for each numbers in Fig,,

	Thermal neutron fluencies [cm^{-2}]	Epithermal neutron fluencies [cm^{-2}]	γ –ray dose [Gy]
①	2.0×10^{12}	3.6×10^{11}	0.61
②	1.5×10^{12}	2.7×10^{11}	0.58
③	1.3×10^{12}	2.3×10^{11}	0.41
④	8.3×10^{11}	1.5×10^{11}	0.43
⑤	6.2×10^{10}	1.1×10^{10}	0.21
⑥	1.6×10^{12}	2.8×10^{11}	0.47
⑦	1.4×10^{12}	2.6×10^{11}	0.60
⑧	1.6×10^{12}	2.8×10^{11}	0.57

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PR7-9 The Study for Development and Application of Tissue Equivalent Neutron Dosimeter

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INTRODUCTION: Recent years, the clinical application of Accelerator-Based Boron Neutron Capture Therapy (AB-BNCT) has been introduced to make significant contributions to treatment for intractable cancer such as glioblastoma multiforme, superficial head and neck cancer, and melanoma in Japan.

In BNCT, the boron (n,α)-reaction of the isotope ^{10}B has a high cross section toward thermal neutrons, and the produced alpha and lithium particles have a short range on the micrometer scale. However, the neutron spectrum always spans a broad energy range, which results in different dose distribution and biological effects in tissue. A radiochromic film (RCF) is one of the most useful dosimetry tools in advantages of high spatial resolution, small energy dependence, tissue equivalence, and self-development without processing in a darkroom[1-3]. The hydrogel material is also expected to use for a patient bolus in clinical radiotherapy[4]. In this work, the authors have developed new nanocomposite hydrogel for BNCT, which is highly expected to use for a patient bolus in clinical BNCT. Moreover, the hydrogel can improve dose distribution as well as can be manufactured arbitrary size and thickness using a 3D printer. Therefore, we have investigated to develop a system that enables dose optimization by optimally modulating the neutron beam for each patient using the hydrogel and RCFs.

EXPERIMENTS: We developed a material jetting 3D printing system with a robust nanocomposite hydrogel. To assess the shielding effect of the 3D printed hydrogel for neutron beam, test slabs were fabricated. The thickness and clay nanoparticles concentration of the test slabs were varied from 3 to 35 mm and 2.5 to 5 wt%, respectively. The irradiation experiment was performed using the standard epithermal neutron irradiation mode of the Heavy Water Neutron Irradiation Facility at Kyoto University Research Reactor (KUR-HWNIF). Gold wires were used to estimate the neutron flux at the entrance and exit plane of the test slabs. Thermo-luminescent dosimeters were used for the estimation of gamma-ray doses. The measured data were normalized by the values at the entrance plane of the test slabs.

RESULTS: The relationship between the thickness of the test slab and the relative doses at the exit plane is shown in Fig. 1a. Both results of neutron flux and gamma-ray dose were firstly a slight increase and then continuous reduce. These phenomena would be affected by the buildup of neutron and gamma-ray doses. The relationship between the clay nanoparticles concentration of the test slab and the relative doses at the exit plane is shown in Fig. 1b. As an overall trend, the relative neutron flux at the exit plane of the test slabs decreased and the relative gamma-ray doses increased with increase in the clay nanoparticles concentration.

The relative neutron flux at the exit plane of test slabs of 3, 5, 10, 20, and 35 mm thickness were 67, 74, 71, 47, and 25%, respectively, and the relative gamma-ray doses were 98, 113, 104, 97, and 82%, respectively. The relative neutron flux at the exit plane of the test slabs decreased and the relative gamma-ray doses increased with increase in the clay nanoparticles concentration.

Our results demonstrated the feasibility of utilizing the 3D printed compensator to modulate the neutron beam intensity for BNCT. This result suggested that the dosimetry using RCFs is essential for a better knowledge of fast neutron flux distribution with the KUR neutron source in previous studies. Moreover, it would be feasible for BNCT dosimetry in medical application.

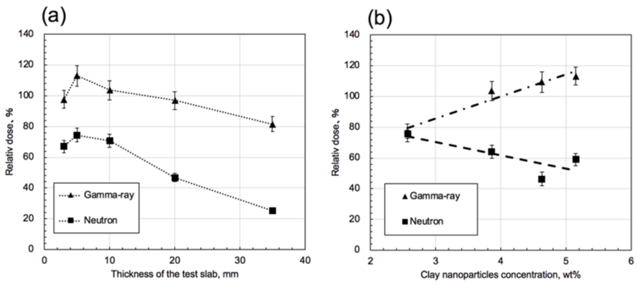

Fig. 1. Relationship between the thickness of the test slab and the relative doses at the exit plane (a), and relationship between the clay nanoparticles concentration of the test slab and the relative doses at the exit plane (b).

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PR7-10 Establishment of beam-quality estimation method in BNCT irradiation field using dual phantom technique (II)

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INTRODUCTION: Research and development into several types of accelerator-based irradiation systems for boron neutron capture therapy (BNCT) is underway [1,2]. Many of these systems are nearing or have started clinical trials. Before the start of treatment with BNCT, the relative biological effectiveness (RBE) for the fast neutrons (over 10 keV) incident to the irradiation field must be estimated.

Measurements of RBE are typically performed by biological experiments with a phantom. Although the dose deposition due to secondary gamma rays is dominant, the relative contributions of thermal neutrons and fast neutrons are virtually equivalent under typical irradiation conditions in a water and/or acrylic phantom. Uniform contributions to the dose deposited from thermal and fast neutrons is based in part on relatively inaccurate dose information for fast neutrons.

The aim of this study is the establishment of accurate beam-quality estimation method mainly for fast neutrons by using two phantoms made of different materials, in which the dose components can be separated according to differences in the interaction cross-sections. The fundamental study of a “dual phantom technique” for measuring the fast neutron component of dose is reported [3].

In 2018, the optimization simulation was performed for the solid phantom. The solid phantoms were made based on the simulation results. The characteristic experiments were performed for these solid phantoms.

METHODS: One solid phantom was made from polyethylene containing natural LiF, and the other solid phantom was made from polyethylene containing enriched ^{6}LiF. The enrichment was 95% for Li-6. Monte Carlo simulations were used to determine the ideal mixing ratio of natural LiF and/or enriched ^{6}LiF in polyethylene.

Two kids of solid phantom were constructed based on the simulation results. One was made from polyethylene containing 30 weight percent natural LiF, the other was made from polyethylene containing 30 weight percent enriched ^{6}LiF.

Experimental characterization of the depth dose distributions of the neutron and gamma-ray components along the central axis was performed at Heavy Water Neutron Irradiation Facility (HWNIF) of KUR using activation foils and thermo-luminescent dosimeters, respectively.

RESULTS: Simulation results demonstrated that the absorbing effect for thermal neutrons occurred when the LiF concentration was over 1%. Finally, the containing ratio of natural LiF and/or enriched ^{6}LiF in polyethylene was determined to be 30 weight percent.

Experiments confirmed that the thermal neutron flux and secondary gamma-ray dose rate decreased substantially however the fast neutron flux and primary gamma-ray dose rate were hardly affected in the ^{6}LiF-polyethylene phantom. Figure 1 shows one of the results for the neutron-flux distributions in the two phantoms. It was confirmed that the dose contribution of fast neutrons was improved from approximately 10% in the LiF-polyethylene phantom, to approximately 50% in the ^{6}LiF-polyethylene phantom.

CONCLUSION: The dual phantom technique using the combination of LiF-polyethylene phantom and ^{6}LiF-polyethylene phantom provides an effective method for dose estimation of the fast neutron component in BNCT. Improvement in the accuracy achieved with the proposed technique results in improved RBE estimation for biological experiments and clinical practice.

We have a plan to perform the biological experiments at KUR-HWNIF, in order to estimate the RBE distribution in these phantoms.

ACKNOWLEDGMENT: This work was supported by JSPS KAKENHI Grant Number JP 16H05237.

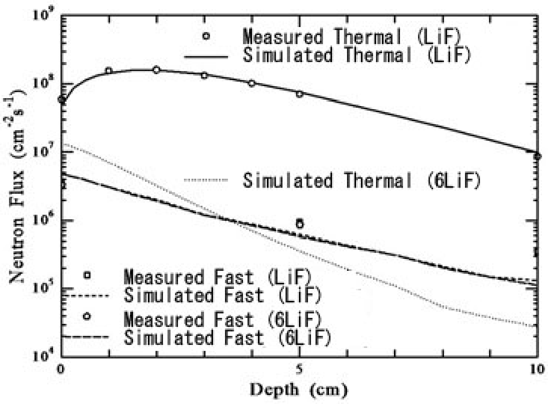

Fig. 1. Comparison between the measured and simulation results for the neutron-flux distributions in the LiF-polyethylene phantom and ^{6}LiF-polyethylene phantom.

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PR7-11 Development of a Prompt Gamma-ray Imaging Detector for Boron Neutron Capture Therapy

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INTRODUCTION: Boron neutron capture therapy (BNCT) is employed to treat cancer cells using a ^{10}B compound and a neutron beam. Basically, the range of the heavy particles, which are produced by the (n, α) reaction between ^{10}B and thermal neutrons, is shorter than the diameter of a cell. A ^{10}B compound accumulates ^{10}B into tumor cells, and into normal cells slightly. To determine the prescript dose during the treatment, it is necessary to measure the ^{10}B concentration in tumor and normal cells in real-time. In addition, it is better to visualize the two-dimensional ^{10}B distribution. At present, it is obtained using a high purity germanium detector with prompt gamma-ray analysis at the Institute for Integrated of Radiation and Nuclear Science, Kyoto University Integrated for Radiation and Nuclear Science (KURNS)[1,2]. However, these procedures are not able to attain the ^{10}B concentration during the irradiation. Thus, a prompt gamma-ray imaging detector system has been developed. It consists of a $LaBr_3$(Ce) scintillator and an 8 x 8 channel multi-pixel photon counter (MPPC), 64 channels amplifiers, a shaper and analog-to-digital converters (ADCs). This paper reports the concept underlying this system and the results of characterizing this system.

EXPERIMENTS: The size of the $LaBr_3$(Ce) scintillator was 50 mm x 50 mm x 10 mm[3]. The scintillator was put in front of an 8 x 8 array MPPC. An MPPC is a type of silicon photomultiplier, and the effective active area of one channel of an MPPC is 6 x 6 mm^2. The outputs of 64 channels were fed to an amplifier unit. The 64 analg outputs were digitalized by ADCs. These digital signals were stored in a PC. Firstly, gain in each channel of the MPPC was adjusted. Secondly, gamma ray spectra from an Na-22 source, which emitted 511 keV gamma rays, were attained in order to confirm that the energy resolution at 511 keV was better than 6.5 %. This is required to discriminate between 478 and 511 keV gamma rays. Finally, gamma ray spectra using samples with 50, 100, 250 and 5000 ppm were obtained by irradiating thermal neutrons at the Kyoto University Reactor neutron guide tube. To discriminate between 478 keV and 511 keV gamma rays, the Gaussian distribution for the two gamma rays was defined.

RESULTS: The average energy resolution at 511 keV gamma rays was approximately 5.4 %. Fig.1 shows the spectrum at the center channel of the MPPC obtained with 50 ppm and three ROIs in the Gaussian distribution for 478 keV gamma rays set up. In addition, the count rate of 478 keV gamma rays and the effect from 511 keV gamma rays were calculated in each ROI. The best ROI was ROI2 because the count rate of 478 keV gamma rays on ROI2 was 0.059 that was sufficient to obtain 100 counts for about 28 minutes that was within common treatment time 1 hour, and the average effect was about 2.9 %. In addition, the linearity of the count rate at 478 keV gamma rays and ^{10}B concentration was confirmed. Furthermore, the two-dimensional boron distribution with 50 ppm on ROI2 was measured in Fig.2.

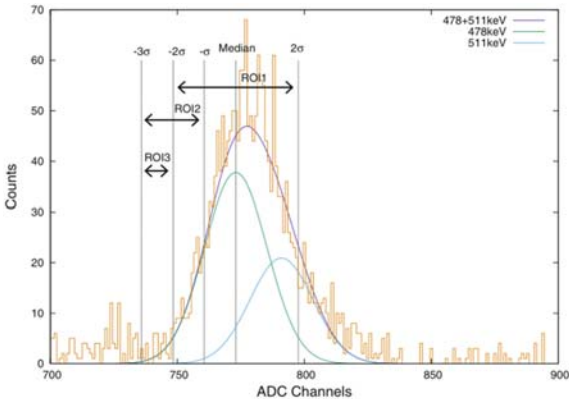

Fig. 1. Gamma rays spectrum at the center channel with 50 ppm.

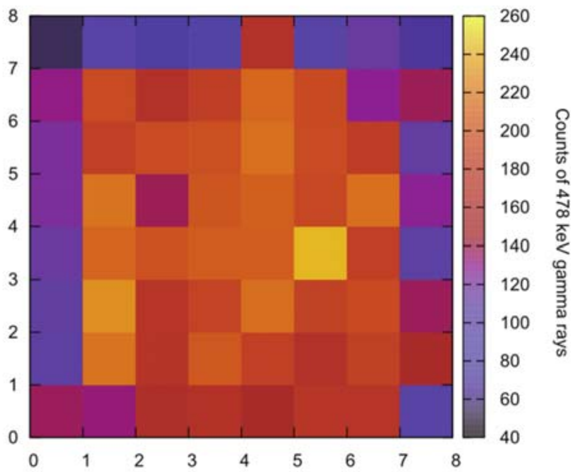

Fig. 2. Two-dimensional boron distribution with 50 ppm.

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30P7-12

PR7-12 Feasibility Study on Ultra-High-Dose-Radiation Monitoring System with Bright-Red Scintillator and Fibers

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INTRODUCTION: Operation of radiation monitor under ultra-high dose is required in several applications such as the Fukushima Daiichi nuclear plants. Since photon counting mode is hard to measure gamma-ray energy due to pile-up under the ultra-high dose, current mode technique is used. Moreover, photo-detectors cannot be operated under such dose condition, Japan Atomic Energy Agency (JAEA) and other groups proposed to read the scintillation photons outside of high-dose condition using optical fibers [1]. The length of optical fiber is expected to be over 100 m, and noise event would be generated excited by gamma rays in the fiber under such high rate dose.

Red-colored emission (~700 nm) is suitable to low transmission loss region of the optical fiber [2]. Ruby (Cr doped α-Al_2O_3) is a one of the candidates for a scintillation probe, because ruby is known to be red-emitter. However, the scintillation decay of ruby is as slow as a few hundred μs, and scintillation signal intensity (pulse height) is regarded to be insufficient.

Recently, we have developed Cs_2HfI_6 (CHI) with a high light output of ~70,000 photons/MeV and moderately fast scintillation decay of ~2 μs [3]. The emission region was observed around 570 - 750 nm (50% of maximum-intensity). The pulse height of CHI is expected to be around 3-4 orders of magnitude higher than that of the ruby scintillator as shown in Figure 2. Thus, this novel scintillator is expected to be applied to the dose monitor.

EXPERIMENTS: We measured the scintillation light output with optical fiber as a function of dose using current mode. The CHI crysal was grown by the vertical Bridgman method in our laboratory, and the crystalline specimen was covered with optical cement.

Scintillation photons for CHI and ruby irradiated with gamma rays from a ^{60}Co source with an activity of around 100 TBq were detected with Si-photodiode (Si-PD) (ThorLabs, SM1PD1A) under the low dose condition using a optical fiber (Fujikura, Low OH type). The signal of the Si-PD was measured with a source meter (Keithlay, 660).

RESULTS: Figure 3 shows the Si-PD current coressponding to light intensity as a fuinction of the dose. Since CHI has brigter emission than ruby, dose dynamic range of the CHI was found to be wider than ruby. In this setup, the length of the fiber is 20 m, while we need to measure the dose with over 100-m lingth fiber in the Fukushima aplication. We found CHI had an advantage to apply into the monitor in Fukushima with longer fiber compared to ruby. Now we also estimate the noise generated in the fiber exctiecd by some dadiation, and we have developed other system using a CCD or other imaging deveises to select the emission wavelength to reject the noise.

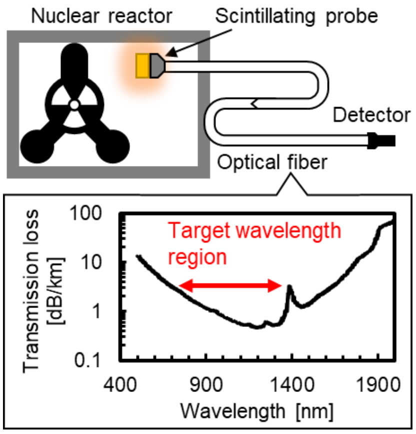

Figure 1. Schematic view of the new radiation detection technique and light transmission loss in an optical fiber.

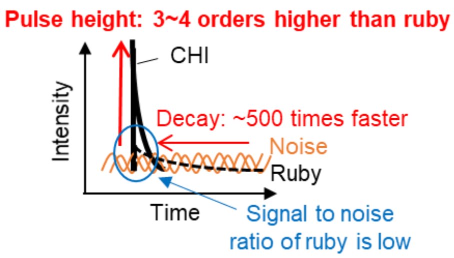

Figure 2. Comparison of time profile of scintillation pulse for CHI and ruby

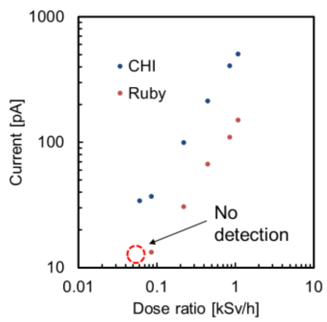

Figure 3. Current from the Si-APD as a function of dose for CHI and ruby

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PR7-13

Establishment of the Imaging Technology of 478 keV Prompt Gamma-rays of Boron-neutron Capture Reaction and the Measurement of the Intensity of the Neutron Field

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INTRODUCTION: Although BNCT is one of the promising cancer treatment methods, we have not yet obtained good method to monitor the treatment effect in real time during BNCT. The main reason is that it is difficult to know precisely both the boron concentration and neutron flux intensity in tumor cells and healthy ones. During BNCT, 478 keV prompt gamma rays are generated by the boron-neutron capture reaction. If we get images of 478 keV gamma rays and know their intensity and generation positions in real time, we can check the treatment effect on BNCT. To get gamma-ray images, there have been proposed several detectors such as SPECT gamma cameras and Compton cameras. In the case of SPECT, a very thick collimators are needed to collimate 478 keV gamma rays. It causes the serious decrease of the detection efficiency and also it would be a main reason of the noise source (511keV) of gamma rays. On the other hand, a Compton camera which does not need to use thick collimator, cannot determine the incident direction of the gamma ray only as a circle due to lack of the direction of recoil electron in Compton scattering, and hence never provide a real and quantitative distributions of Boron in body.

As the detector which overcome the shortcomings of two detectors, we have been developing electron tracking Compton cameras (ETCCs). ETCCs are advanced Compton cameras which take the information of the recoil direction of electrons and can uniquely determine the arrival direction of sub-MeV/MeV gamma ray event by event. An ETCC is the complex detector of two sub detectors: a time projection chamber TPC (Compton scatterer and recoil electron detector) and a scintillation camera (scattering gamma-ray absorber). The detail structure of ETCCs are written in [1][2].

EXPERIMENTS: Previously we developed a small ETCC which has a scintillation camera constructed with 576 pixelated GSO scintillators, whose pixel size is 6 mm x 6 mm (26 mm in high), and a TPC whose size is 100 cm^2 (15 cm in high) [2]. To improve the effective area, we developed a prototype middle size ETCC which has the same number of scintillators and a TPC whose size is 400 cm^2 (20 cm in high). We also developed a new middle size ETCC which has a TPC whose size is ~330 cm^2 (20 cm in high). We succeeded in the reduction of the size of ETCC. The total size of the new ETCC is roughly the same as that of the previous small ETCC.

As the first step of the imaging performance test for 478 keV prompt gamma rays, we performed an experiment at KURRI. We irradiated a Boron target (high concentration Boron solution) with thermal neutron beam from Kyoto University Research Reactor (KUR). During the irradiation, we set the ETCC and a germanium (Ge) detector near the target. The exposure time is about an hour. The left part of the Fig.1 shows the photograph and schematic of the experiment.

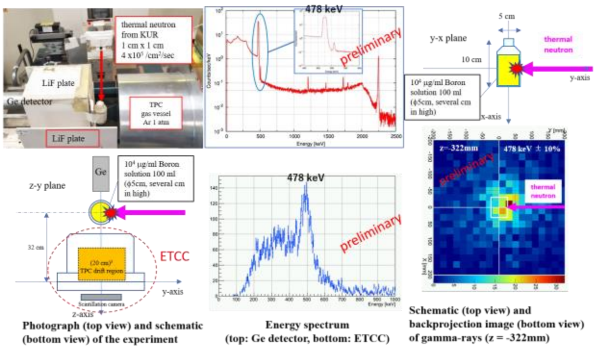

Fig.1: Experimental setup and the results. Left panel: photo and schematic of the experiment. Center panel: Spectra of the gamma-rays taken by Germanium detector and the ETCC. Right panel: schematic and back-projection image of 478 keV gamma-rays.

RESULTS: The center panel of Fig.1 shows energy spectra taken by the Germanium detector and ETCC. As shown in the spectrum of germanium detector, 478 keV gamma-rays have an overwhelmingly larger number of 511 keV gamma-rays. In the ETCC spectrum, we can see a clear peak of 478 keV gamma-rays. The right part of Fig.1 shows the schematic view and the ETCC 478 keV gamma-ray back-projection image on z = -322 mm plane (z axis is defined in Fig.1). In the back-projection image, the light point in the image corresponds to the intersection point of the neutron beam and the surface of the Boron solution container. From the results, we succeeded in imaging 478 keV prompt gamma-rays of Boron neutron capture reaction with ETCC. In the future prospects, it is needed to try the measurement test in the high dose condition and the separation of 478 keV and 511 keV gamma-ray events.

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PR7-14 Feasibility study for establihing quality assurance and quality control for radiation field in boron neutron capture therapy

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INTRODUCTION: Boron neutron capture therapy (BNCT) has been performed using the research nuclear reactor as neutron sources [1], [2]. However, recent researches have indicated that accelerator-based neutron sources can be employed in BNCT. Therefore, in order to evaluate the efficacy of BNCT in clinical oncology, the accelerator-based neutron source is installed into the hospital, such as National Cancer Center Hospital, Tokyo, Japan, and Southern TOHOKU General Hospital, Fukushima, Japan [3], [4], [5], [6]. Compared to the research nuclear reactor, the size of facility in the accelerator-based neutron source is small. In the radiation therapy, quality assurance (QA) and quality control (QC) is important to validate the treatment quality. In BNCT, the radiation field has both neutrons and gamma-rays. Hence, methods of the quality assurance (QA) and the quality control (QC) for the radiation field in BNCT have to consider its property. Nevertheless, the detector for neutron and gamma-ray is not enough small to be equipped into the accelerator-based neutron source. The purpose of this study is to establish the QA and QC methods for radiation field in BNCT with considering the property in the accelerator-based neutron source.

EXPERIMENTS: In order to establish the method for QA and QC, three types of films were examined. Two of those consisted of thermo-luminescence phosphor. After radiation exposure, with heating the film, output can be measured. Signal intensity was measured as the amount of the luminescence. The other was EBT3 film (manufactured by Ashland). The experiment of neutron irradiation was performed in Kyoto University Research Reactor (1 MW). Pictures of these experiments were shown in figure 1. TLD detectors for measuring gamma-ray and a gold foil for measuring the neutrons were placed on each of the films, and the neutrons and the gamma-rays were then delivered to those films. The delivered time was set to 30 s, 1 min, and 2 min.

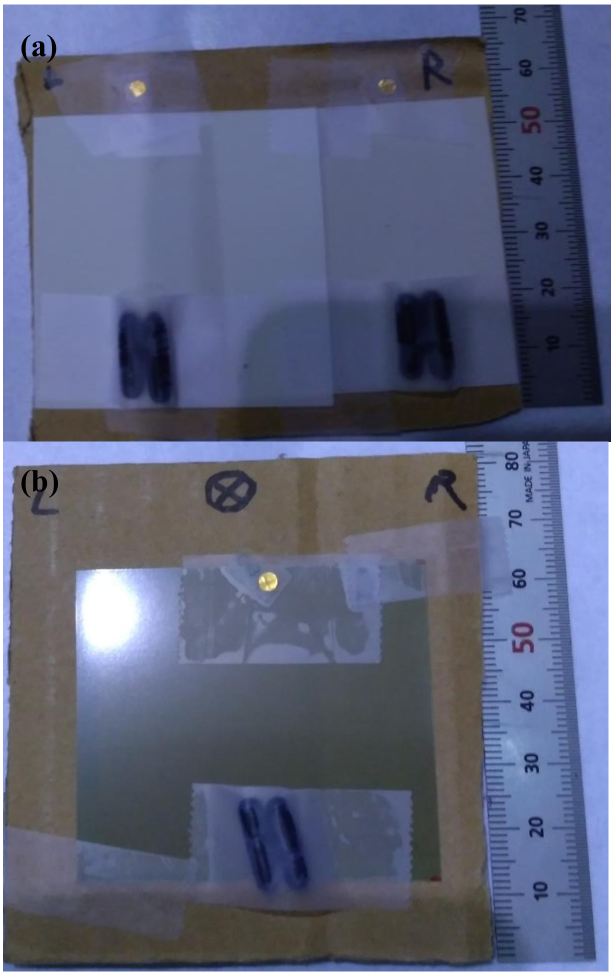

Fig.1. The picture of each experiment. (a): two films consist of thermo-luminescence phosphor. (b): EBT3 film

RESULTS: The output depended on the delivered time in each film. The outputs of three films represented different dose.

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Patient-Position Monitoring by Using Kinect Sensor for Boron Neutron Capture Therapy

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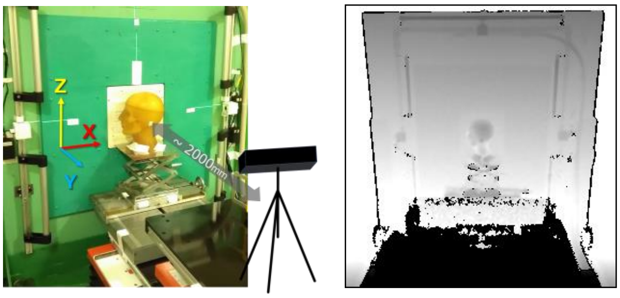
INTRODUCTION: In boron neutron capture therapy (BNCT) irradiations carried out at Kyoto University Research Reactor, sitting position has been applied in many cases, considering flexibility of patient positioning and structural restriction of an irradiation facility. In some cases, there is difficulty in reproducing a patient position determined by a treatment planning process, which is related to a patient set-up error. Also, the sitting position is sometimes unstable, resulting in displacement from an initial set-up position during an irradiation period, which is related to patient motion. These set-up error and motion cause uncertainty in estimation of delivered dose.

Aiming to improve the dose estimation accuracy, we have been developing a patient-position monitoring system using a Kinect sensor: a real-time range sensing device [1]. This report describes measurement of patient position displacement by using a Kinect sensor.

EXPERIMENTS: Measurement using a human head phantom was performed in a treatment room at KUR heavy water neutron irradiation facility, which is usually used for patient positioning prior to BNCT irradiation. The experimental setup is shown in Fig. 1. The head phantom was placed on a XYZ stage. A Kinect sensor was pointed to the phantom from the downstream of beam axis at a distance of about 2m. The phantom was moved to each of XYZ directions shown in Fig. 1(a) in a range from 3 to 10mm, and 2D range image was acquired with the sensor at each of the positions. An example of the 2D range image is shown in Fig. 1 (b).

Procedure of displacement estimation is described in Fig. 2. The 2D range image at each position was converted to point cloud data format, then the data points in a background region were masked. Displacement was estimated according to the transformation vector to the original position, which was obtained by making registration of point clouds. The VC++, Kinect for Windows SDK v2 and PCL (Point Cloud Library) ver. 1.8.1 [2] were used for handling the 2D range images and point cloud data. The registration of the point clouds was performed by means of Iterative Closest Point (ICP) algorithm [3] implemented in the PCL library.

RESULTS: The estimated displacements are shown in Table 1. Displacements in a range of ~10mm were estimated with good accuracy in directions normal to the y axis. However, there were large discrepancy in the y axis along the sensor-to-object direction. Ambient IR rays can be considered a cause of fail in measurement because a Kinect sensor uses an IR time-of-flight method to measure a range to an object. Especially, the intense IR rays from mercury lamps attached on the ceiling of the reactor room were considered to affected largely on the measurement performance of the Kinect. Shielding the ambient IR can be effectively work to improve the results.



(a) (b)

Fig. 1. Experimental setup for position measurement using a head phantom: (a) arrangement of a phantom and a Kinect sensor, shown with definition of spatial coordinate, (b) an example of range image acquired with the sensor.

Fig. 2. Procedure of displacement estimation. Point cloud presented with white dots is corresponding to original position.

Table 1. Estimation results of phantom displacement measured with the Kinect sensor.

Displacement (mm)	X		Y		Z	
	+3	+10	+3	+10	-3	-10
Estimated (mm)	+2.5	+10	+0.7	+3.5	-7	-9

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