# PR12 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 first year of this research project, 2017, the fundamental characterization for the respective methods was performed mainly using Heavy Water Neutron Irradiation Facility of KUR, and the direction for the advance was decided.

**RESEARCH SUBJECTS:**

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

**ARS-1 (29P12-1):** Establishment of characterization estimation method in BNCT irradiation field using Bonner sphere and ionization chamber. (Y. Sakurai, R. Uchida, T. Takata, H. Tanaka, S. Shiraishi, N. Ko, K. Okazaki, T. Kawamura, M. Sato, K. Akita, and M. Suzuki)

**ARS-2 (29P12-2):** Basic study on new type of neutron spectrometer for epi-thermal energy region. (A. Uritani, N. Suda, K. Watanabe, S. Yoshihashi, A. Yamazaki, H. Shimizu, and Y. Sakurai)

**ARS-3 (29P12-3):** Improvement of SOF detector system for long-term stability. (M. Ishikawa, Y. Murayama, K. Baba, Y. Sakurai, and M. Suzuki)

**ARS-4 (29P12-4):** Response of a commercial CsI detector for the self-activation method in BNCT field. (A. Nohtomi, R. Kurihara, G. Wakabayashi, Y. Sakurai, and T. Takata)

**ARS-5 (29P12-5):** Neutron beam quality and dose measurement of the Kyoto University Research Reactor using microdosimetric technique. (N. Ko, S. Endo, K. Tanaka, T. Kajimoto, M. Takada, T. Takata, Y. Sakurai, and H. Tanaka)

**ARS-6 (29P12-6):** Study for microdosimetry using silicon-on-insulator microdosimeter in the BNCT irradiation field. (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 (29P12-7):** Estimation of dose resolution by gel detector for BNCT. (R. Maruta, K. Tanaka, Y. Murakami, Y. Sakurai, T. Kajimoto, H. Tanaka, T. Takata, and S. Endo)

**ARS-8 (29P12-8):** Study on the development of neutron fluence distribution measurement device using thermoluminescence of the ceramic plates. (K. Shinsho, S. Yanagisawa, Y. Koba, G. Wakabayashi, and H. Tanaka)

**ARS-9 (29P12-9):** The study for development and application of tissue equivalent neutron dosimeter. (M. Oita, T. Kamomae, T. Takada, and Y. Sakurai)

**ARS-10 (29P12-10):** 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 (29P12-11):** Establishment of beam-quality estimation method in BNCT irradiation field using dual phantom technique. (Y. Sakurai, T. Takata, H. Tanaka, N. Kondo, and M. Suzuki)

**ARS-12 (29P12-12):** Development of real-time dose monitor using prompt rays imaging detector for boron neutron capture therapy. (H. Tanaka, K. Okazaki, T. Takata, Y. Sakurai, and M. Suzuki)

**ARS-13 (29P12-13):** 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, and H. Tanaka)

**ARS-14 (29P12-14):** Basic study for QA/QC in BNCT irradiation field. (S. Nakamura, T. Nishio, H. Okamoto, A. Wakita, S. Ito, J. Itami, H. Igaki, M. Munechika, S. Tanaka, Y. Sakurai, H. Tanaka, T. Takata, and M. Suzuki)

**ARS-15 (29P12-15):** Patient-position monitoring system for BNCT irradiation. (T. Takata, H. Tanaka, Y. Sakurai, and M. Suzuki)

For ARS-14, no results were obtained because no machine time were used due to the short of operation time of KUR and the schedule of the collaborators. So, the report of this research subject is not appeared.

29P12

# PR12-1 Establishment of Characterization Estimation Method in BNCT Irradiation Field using Bonner Sphere and Ionization Chamber

Y. Sakurai, R. Uchida[1], T. Takata, H. Tanaka, S. Shiraishi[1], N. Ko[1], K. Okazaki[1], T. Kawamura[1], M. Sato[1], K. Akita[1,2] and 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]. The development of this remote-changeable Bonner-sphere spectrometer is reported.

**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.

**RESULTS:** The optimized structure was selected by HIS as follows: three sphere shells such as 13, 18 and 20 cm in diameter, and three liquid moderators such as pure water, 0.028-wt% boron acid solution and 0.7-wt% boron acid solution, as shown in Fig.1.

**CONCLUSION:** We have a plan to make the remote-changeable Bonner-sphere spectrometer, based on the optimization result. Additionally, we have a plan to perform the spectrometry experiments at Kyoto University Reactor (KUR), etc., in order to confirm the efficacy of this spectrometer.

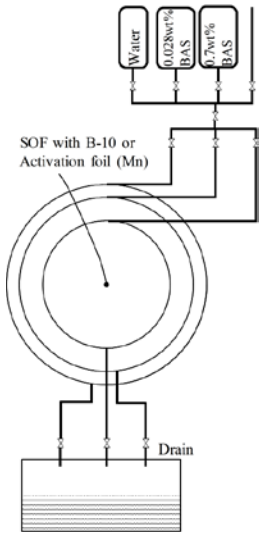


Fig. 1. Optimization for the structure of the remote-changeable Bonner-sphere spectrometer.

**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).

29P12-1

## PR12-2 Basic Study on New Type of Neutron Spctrometer for Epi-thermal Energy Region

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**INTRODUCTION:** Boron neutron capture therapy (BNCT) is one of the promising treatment methods for cancers such as brain tumors. The principle of BNCT was proposed by Locher [1] in 1939. In recent years, an accelerator-driven neutron source has been developed due to its simplicity of management. The C- BENS [2], the first accelerator-driven BNCT facility in the world, built at the Kyoto University Research Reactor Institute. This facility has already been in the clinical trial phase. 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 an optical fiber type neutron detector, which consists of a small Eu:$LiCaAlF_6$ scintillator, an optical fiber light guide and photomultiplier tube, for this purpose [3].

In this study, we are developing a new neutron energy spectrometer especially for epithermal neutrons using the optical fiber type detector. The spectrometer consists of some detectors surrounded by relatively small size moderators with various size and/or covered by various resonance foil absorbers. The small size moderators emphasize the energy response for epi-thermal region. In addition, the subtraction method using the resonance absorption foils also improves the epi-thermal energy response. In this paper, we evaluated the energy response of the designed detectors using Monte Carlo simulation code PHITS. We, additionally, carried out the detector tests at the Heavy Water Neutron Irradiation Facility of Kyoto University Research Reactor.

**RESULTS:** Figure 1 shows the energy responses of the designed detectors. As shown in Fig. 1, some detectors show narrow dips in the response function, corresponding to resonance absorption of each foil absorber. The subtraction method can extract narrow band energy response due to these dip components.

We fabricated the proto-type detector using the optical fiber type detector surrounded by a 5 mm thick polyethylene moderator and covered by a tungsten resonance filter and a cadmium thermal neutron shielding. The sensitivity of the fabricated detector was evaluated to be the order of $10^{-7}$ $cm^2$. This sensitivity level is adequate for characterization of the BNCT irradiation fields.

**REFERENCES:**

[1] G. L. Locher: Am. J. Roentgenol **36**, 1 (1936).

[2] H. Tanaka, Y. Sakurai, M. Suzuki, S. Masunaga, T. Mitsumoto, K. Fujita, G. Kashino, Y. Kinashi, Y. Liu, M. Takada, K. Ono and A. Maruhashi, Applied Radiation and Isotopes, **69**, 1642 (2011).

[3] K. Watanabe, Y. Kawabata, A. Yamazaki, A. Uritani, T. Iguchi, K. Fukuda, T. Yanagida , Nuclear Instruments and Methods in Physics Research Section A , **802**, 1 (2015)

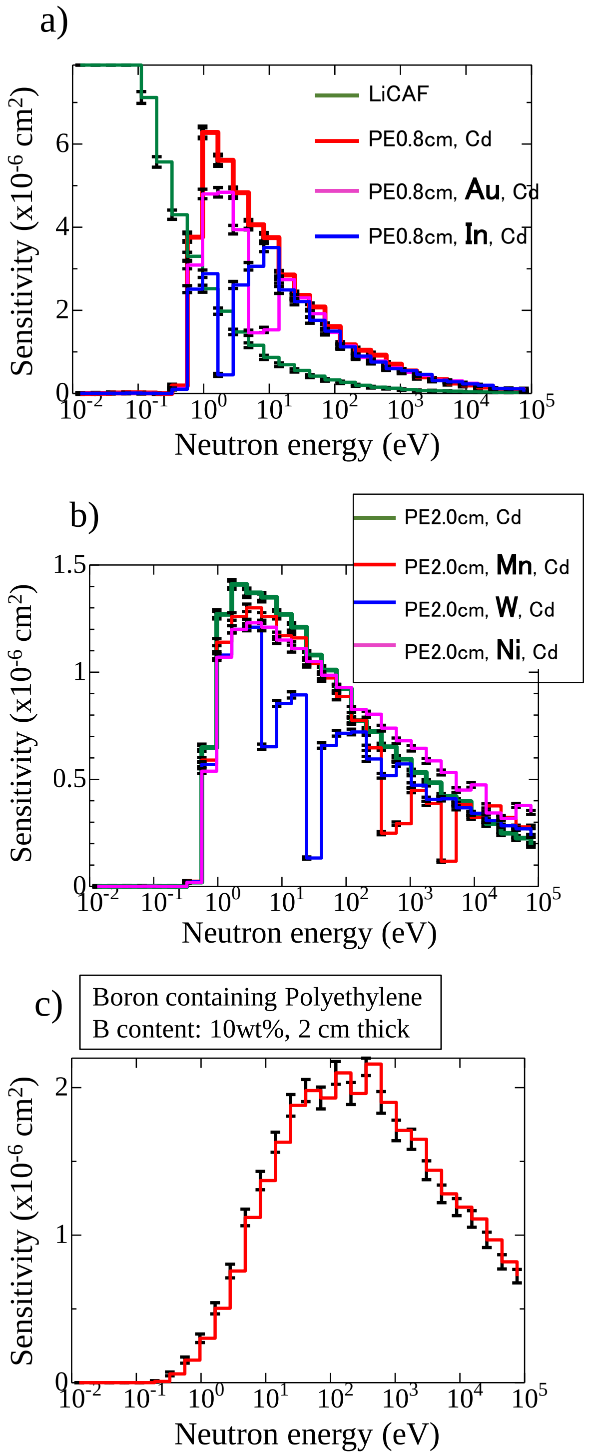


Fig. 1. Energy responses of designed detectors. a) Bare detector and detectors with 8 mm thick PE moderator and some resonance absorbers. b) detectors with 20 mm thick PE moderator and some resonance absorbers. c) detectors with 20 mm thick B-PE moderator.

# PR12-3 Improvement of SOF Detector System for Long-term Stability

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**INTRODUCTION:** In boron neutron capture therapy, absorbed dose is evaluated from boron concentration and thermal neutron flux in the tumor area. Conventionally, since gold activation method was used as an evaluation method of thermal neutron flux, absorbed dose could not be evaluated in real time. Therefore, our laboratory has developed a SOF detector (Scintillator with Optical Fiber Detector) to measure thermal neutron flux in real time [1, 2] and reported the usefulness of real-time measurement in clinical practice [3].

As shown in Fig. 1, the SOF detector consists of a small amount of plastic scintillator, a plastic optical fiber, a photo-multiplier tube, a charge pre-amplifier, a discriminator and a counter. The plastic scintillator BC490 is attached to the tip of the plastic optical fiber. A Small amount of LiF powder (enriched 95% $^{6}$Li) is then painted over the plastic scintillator. The reactions between 6Li nuclei and thermal neutrons emit charged particles (alpha and triton) which produce scintillation photons in the plastic scintillator. The photon signals are relayed through an optical fiber to the Photon Counting Unit (Hamamatsu H7155) then converted into 30 nsec-width TTL pulse [1]. A paired-SOF method was used for gamma-ray compensation [2].

The SOF detector can measure thermal neutron flux in real time with very wide dynamic range, however, there is a report that output is decreased by long-term measurement under high thermal neutron flux [4]. Therefore, the purpose of this research is to improve the SOF detector system for enabling long-term stable measurement.

**EXPERIMENTS:** In this study, it is necessary to know the long-term change in measurement performance of the SOF detector. However, the research reactor at Kyoto University Reactor is not available for a long time, we have to calibrate the SOF detector. As shown in Fig. 2, pure thermal neutrons were irradiated to the SOF detector using the E-3 neutron guide tube facility, and conversion coefficients were calculated by comparing a thermal neutron flux evaluated by the gold foil.

**RESULTS:** A

A thermal neutron flux of the neutron beam from the E-3 neutron guide tube was evaluated using a 50 μm-thick gold foil. The evaluated thermal neutron flux at measured position was $2.65\times10^{5}$ [n/cm$^2$/s], and the total thermal neutron fluence was calculated as $2.55\times10^{8}$ [n/cm$^2$]. From this thermal neutron fluence, the conversion coefficient of the SOF detector was determined to be $2.41 \times 10^{3}$ [n/cm$^2$/counts]. For the future work, we will monitor the conversion coefficient change according to high thermal neutron flux irradiation to the SOF detector.

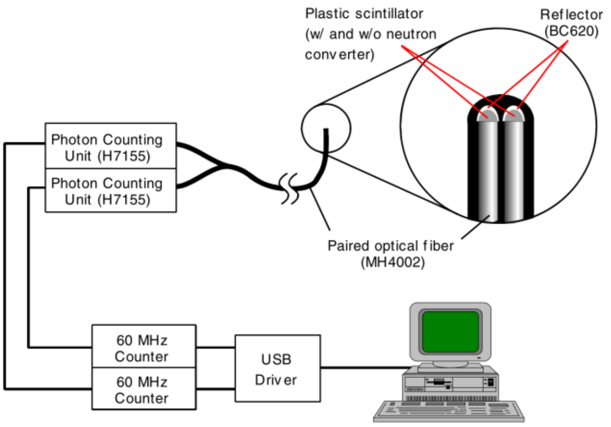


Fig. 1. Schematic illustration of SOF detector system. The SOF detector system consists of a small amount of plastic scintillator, a plastic optical fiber, a photo-multiplier tube, a charge pre-amplifier, a discriminator and a counter.

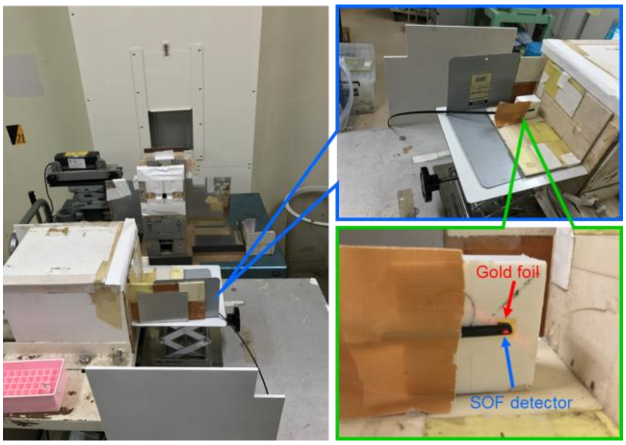


Fig. 2. Calibration geometry for the SOF detector using pure thermal neutrons from the E-3 neutron guide tube facility. The conversion coefficients were calculated by comparing a thermal neutron flux evaluated by the gold foil.

**REFERENCES:**

[1] M. Ishikawa *et al.*, Appl. Radiat. Isot., **61** (2004) 775–779.

[2] M. Ishikawa *et al.*, Nucl. Instr. and Meth., A **551** (2005) 448–457.

[3] M. Ishikawa *et al.* Radiat. Oncol., **11:105** (2016) DOI: 10.1186.

[4] M. Komeda *et al.* Appl. Radiat. Isot., **67** (2009) S254-S25.

# Resopnse of a Commercial CsI Detector for the Self-activation Methodin BNCT Field

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**INTRODUCTION:** In our previous studies [1 - 5], the self-activation of iodine-containing scintillator had been successfully applied for detecting photo-neutrons around a high-energy X-ray radiotherapy machine. Absolute evaluation of neutron intensity is possible by the so-called activation analysis with online measurement. Especially, CsI scintillator is only slightly hygroscopic, and the light output is easily read out by a PD that does not need high-voltage power supply. The self-activation technique has been originally developed for the high sensitive neutron detection at a weak neutron field. In the present work, the applicability of commercially available CsI detectors were studied at rather intense neutron filled like BNCT one of KUR irradiation facility for short time irradiation.

**EXPERIMENTS:** Two types of CsI detectors, Hamamatsu C12137 and Horiba PA1100 (Radi) were irradiated at Irradiation Rail Device of the Heavy Water Neutron Irradiation Facility with OO-0000F mode (1MW) [6]. The detectors were put at 1 m from the Bismuth Surface (thermal neutron flux : 5.1 x $10^7$ n/cm$^2$/s) during about 2 seconds, or at the entrance position of Rail Device during about 130 seconds. Two of Radis were covered with 1 mm-t Cd sheet or 5 mm-t $B_4C$ silicon rubber sheet (70%).

**RESULTS:** Figure 1 shows a typical energy spectrum observed by C12137 just after the termination of 2 seconds irradiation at 1 m from the Bismuth Surface. In addition to continual spectrum of $^{128}$I beta ray, a conversion electron peak of $^{134m}$Cs (~140 keV) and a characteristics X-ray peak of $^{128}$Te (~30 keV) can be clearly seen.

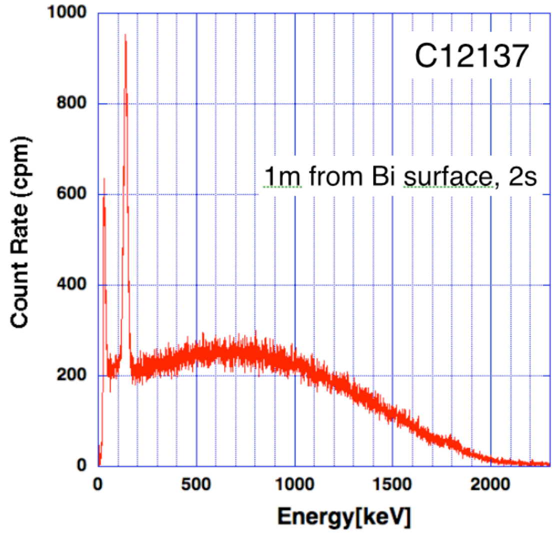


Fig. 1. Typical energy spectrum of CsI self-activation observed by C12137.

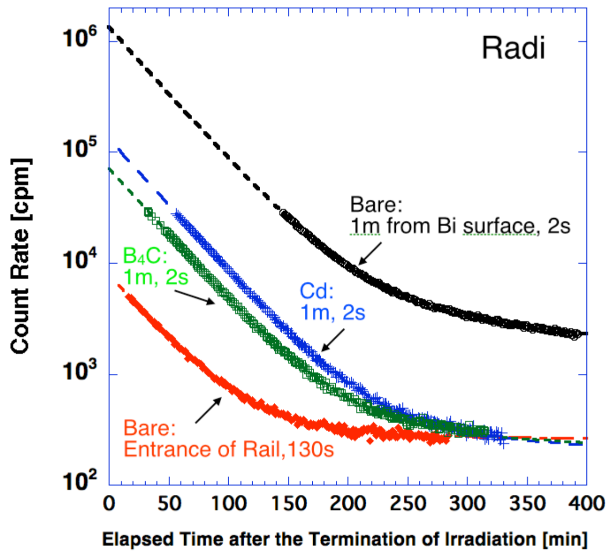


Fig. 2. Decay curves of count rate obtained by Radi for different irradiation conditions.

Decay curves of count rate obtained by Radis are plotted in Fig. 2. Acquired data for every 1 minute through bluetooth interface were fitted well with a combination of $^{128}$I ($T_{1/2}$=25min), $^{134m}$Cs ($T_{1/2}$=174min) and constant B.G. components. Then, initial count rate of $^{128}$I were calculated to evaluate the initial activities. When Raids were put at 1m from the Bismuth Surface and irradiated about 2 seconds, the initial activity was 2.58 x $10^4$ [Bq] for Bare (un-covered) Radi, 2.54 x $10^3$ [Bq] for the Radi with Cd cover and 1.37 x $10^3$ [Bq] for the Radi with $B_4C$ cover. When a Bare (un-covered) Radi was put at the Entrance of Rail Device and irradiated about 130 seconds, the initial activity was evaluated to be 1.43 x $10^2$ [Bq] .

From the results mentioned above, it has been confirmed that those commercially available CsI detectors work properly for short time irradiation of intense neutron beam and are applicable to the self-activation method in BNCT field.

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

**REFERENCES:**
[1] G.Wakabayashi *et al*., Radiol. Phys. Technol., **8** (2015) 125-134.
[2] A.Nohtomi and G.Wakabayashi, Nucl. Instr. and Meth., **A800** (2015) 6-11.
[3] A.Nohtomi *et al*., JPS Conf. Proc., **11** (2016) 050002.
[4] A.Nohtomi *et al*., Nucl. Instr. and Meth., **A832** (2016) 21-23.
[5] S.Honda *et al*., Nucl. Instr. and Meth., **A871** (2017) 148-153.
[6] Y.Sakurai and T.Kobayashi, Nucl. Instr. and Meth., **A453** (2000) 569-596.

# PR12-5 Neutron Beam Quality and dose Measurement of the Kyoto University Research Reactor Using Microdosimetric Technique

N. Ko, S. Endo[2],K. Tanaka[2], T. Kajimoto[2], M. Takada[3], T. Takata[1], Y. Sakurai[1] and H. Tanaka[1]

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*3Applied Physics, National Defense Academy*

**INTRODUCTION:** The aim of BNCT is the local energy deposition of high LET particles to the target area. In addition to alpha particles and lithium ions generated from the boron-10 and thermal neutron interaction, protons and gamma rays are also present in the BNCT field. Microdosimetry is an effective tool to measure radiation in a mixed field. Using this technique, it is possible to derive the relative contribution of each radiation component. A tissue equivalent proportional counter (TEPC) measures energy deposition in a simulated micrometer scale volume comparable to that of a living cell. The TEPC uses material and gases that are essentially equivalent to human tissue in chemical composition. This paper presents the radiation beam quality measurements of the thermal and mixed mode irradiation of Kyoto University Reactor (KUR) that is used for both clinical and non-clinical studies, respectively.

**EXPERIMENTS:** The neutron single event spectrum was measured using the TEPC (LET 0.5 inch chamber, Far West Technology Inc.). To simulate a 1 μm diameter sphere, methane-based tissue equivalent gas was filled at a pressure of 74.5 hPa. The KUR heavy water neutron irradiation facility was operated at thermal, epithermal and mix mode irradiation. The TEPC was placed on top of the treatment couch in the center of the field, as shown in figure 1. Measurements were performed free-in-air and the beam on time for each irradiation was approximately 1 hour. Microdosimetric single event spectra was measured and the event and dose frequency spectra were calculated using the lineal energy, y, which is the deposited energy in an active volume of diameter divided by the mean chord length. Particle and Heavy Ion Transport code System (PHITS) version 2.88 was also used to calculate the microdosimetric spectrum.

**RESULTS:** Figure 2 shows the microdosimetric spectrum of the mix irradiation mode of KUR measured in air using the TEPC. The electron and proton edge were clearly visible at approximately 15 keV/μm and 100 keV/μm, respectively. Figure 3 shows the dose distribu-tion spectrum measured with the TEPC. The major dose contributor was the proton component, having a lineal energy between 20-100 keV/μm, followed by the carbon ions at approximately 300 keV/μm.The PHITS calculation showed similar results to the measured data. The y values below 10 keV/μm were not obtained due to the high electrical noise of the system. Therefore, all of the gamma components could not be measured. Similar spectrum was obtained with the epithermal irradiation mode and comparison was made with previous experiment performed by Endo et.al [1].

Fig. 1. Image of the experimental set up inside the KUR irradiation room.

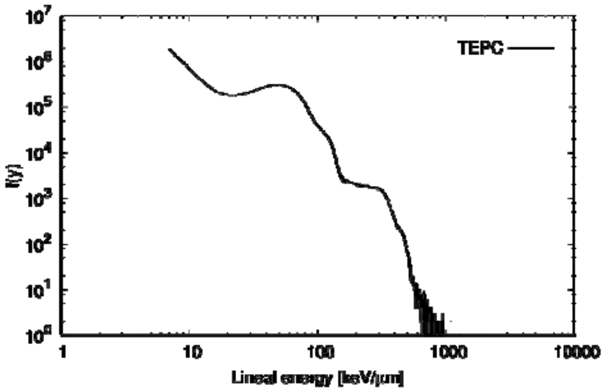


Fig. 2. Microdosimetric single event spectrum of f(y) free in air for mix irradiation mode.

**REFERENCE:**
[1] S. Endo *et al.*, Radiat. Prot. Dosimetry, **110** (2004) 641-644.

PR12-6

# Study for Microdosimetry using Silicon-on-insulator Microdosimeter in the BNCT Irradiation Field

Y. Sakurai, N. Ko[1], R. Uchida[1], T. Takata, H. Tanaka, T. L. Tran[2], J. Davis[2], 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 microdosimeter in BNCT [3], developed by University of Wollongong (UOW).

**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.
The performance of the microdosimeter was studied using Monte Carlo simulation. 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 Kyoto University Reactor (KUR) using both thermal and epithermal irradiation modes were used in this study.

**RESULTS:** A thinner boron converter resulted in more reaction particles reaching the sensitive volume of the detector. Approximately double the number of particles reached the detector for a 1-μm thick boron converter as compared to a 0.5 mm thick boron converter. A peak at 120 keV/μm was observed with both the 0.5mm and 1μm boron converter and a peak at 200 keV/μm was observed with no boron converter. A peak in the no converter spectrum arises from the boron p+ dopant in the device.
Figure 1 shows the microdosimetric spectrum obtained from the bridge microdosimeter for the KUR epithermal beam. The range of the alpha particles produced from BNCT is approximately 5 μm. For a 10-μm thick detector, the alpha particle will come to a full stop inside the sensitive volume, resulting in an inaccurate lineal energy deposition. A 2-μm thick detector was simulated and tested. Results showed the lineal energy deposition was improved with the use of the 2-μm thick sensitive volume detector.
Simulation results showed that the thermal irradiation mode seemed the most appropriate mode to perform the measurements. This was due to the high thermal neutron flux, which resulted in high production of reaction particles, and lower epithermal and fast neutron flux, which resulted in lower recoil silicon particles produced.

**CONCLUSION:** The microdosimetric spectrum showed each particle can be separated by the use of the lineal energy. The simulation results show that this microdosimeter can be utilized as an effective tool for dosimetry in BNCT field. Experimental validation is planed using KUR.

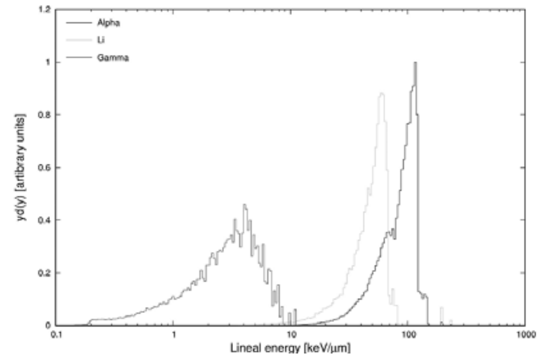


Fig. 1. Microdosimetric spectrum of the KUR epithermal beam generated by PHITS.

**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] L. T. Tran *et al*., IEEE Trans. Nucl. Sci. **62** (2015) 3027-3033.

29P12-6

# Estimation of dose Resolution by Gel Detector for BNCT

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**INTRODUCTION:** Evaluation of dose is required for quality assurance in the irradiation field used for boron neutron capture therapy. This study investigated the use of the MAGAT polymer gel detector in the QA by esti-mation of dose and position resolution.

**EXPERIMENTS:** The response of the MAGAT to dose was measured. The irradiation was performed by using Co gamma rays irradiation facility at Hiroshima University. The transverse relaxation rate ($R_2$) was determined using a 0.3T MRI scanner (AIRIS Ⅱ comfort, Hitachi Medical Corp.) with a standards head coil. It is assumed that the difference in dose can be distinguished if signal intensity differs by 3 times standard deviation (SD). The parameters ‘a’ and ‘b’ in Fig.1 were estimated by measuring $R_2$ for regions with varied diameters. ‘c’ in Fig.1 was estimated from distance of $R_2$ decrease from the gel to its outside.

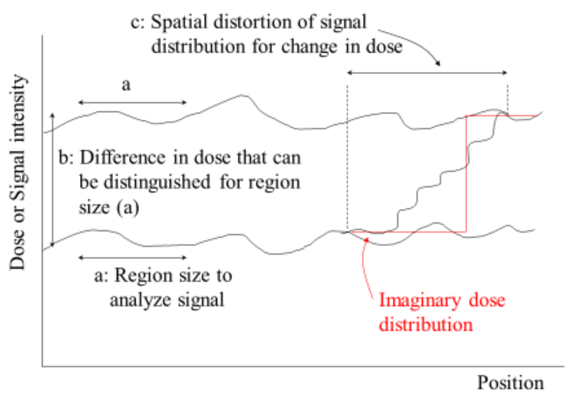


Fig. 1 Schematic drawing of dose and position resolution to be estimated.

**RESULTS:** As the diameters to be analyzed get smaller, dose resolution tends to decrease. On the other hand, one of the examples that dose resolution could be varied largely at smaller diameters is shown in Fig 2. The diameter of ‘a’ is required to be 3pixel (3.75mm) or more to distinguish the difference in dose.

Figure 3 shows the decrease of dose resolution associated with the decrease in $R_2$ at 3pixel diameters. ‘b’ in Fig.1 is estimated to be about 10-30%.

Figure 4 shows the decrease in $R_2$ through the gel (-about 6mm) and periphery (about 6mm-). It is estimated that about 80% decrease in $R_2$ requires the distance by 2pixel (2.5mm, from 5mm to 7.5mm), and 6pixel (7.5mm, from 5mm to 12.5mm) is required for the decrease in $R_2$ from the gel to periphery region (considered as a convergence value). Therefore, ‘c’ is estimated to be about 2.5-7.5mm.

In conclusion, when the irradiation field has the region with low dose or high dose , it is estimated that the difference in dose by 10-30% can be distinguished for the region size of 8.75-18.75mm (7-15pixel).

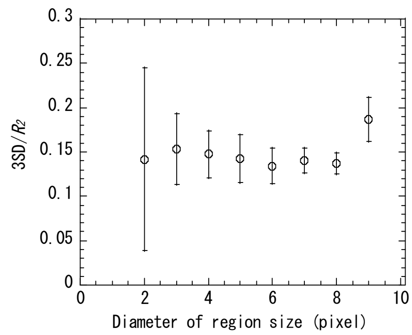


Fig. 2 Dose resolution dependence on diameter of region size to be analyzed signal.

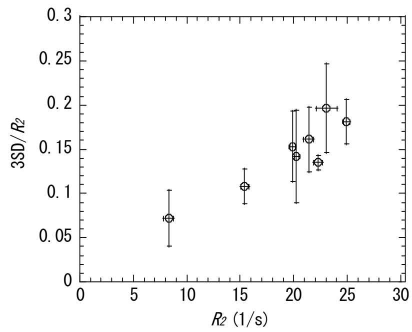


Fig. 3 Dose resolution dependence on signal intensity at 3pixel diameters.

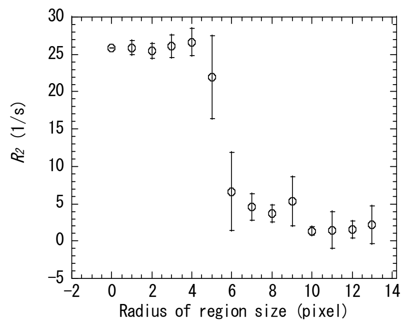


Fig. 4 Change in signal intensity through gel and periphery.

**PR12-8 Study on the Development of Neutron Fluence Distribution Measurement Device Using Thermoluminescence of the Ceramic Plates**

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**INTRODUCTION:** BNCT (boron neutron capture therapy) is a next-generation cancer treatment that uses a boron compound and neutron irradiation to kill cancer cells selectively. Neutron fluence distribution measurement and the absorbed dose distribution measurement of tumors are required for planning improvement of the treatment precision of BNCT. We have studying TL gamma-ray detectors that primarily use Cr doped $Al_2O_3$. [1,2] Luckily, we have also discovered that it has sensitivity toward neutrons, but now wish to increase this sensitivity. As boron has a high neutron cross section, we co-doped $Al_2O_3$ with Cr and B, in order to generate a higher sensitivity. We investigated the TL glow curves and the dose linearity for neutron and gamma mixed fields.

**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%, $Na_2O$ < 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 mm × 10 mm × 1 mm. The concentration of $Cr_2O_3$ in the present study was 0.05 wt% and the concentration of and $^9B_2O_3$ was 0.1 wt%. The assumed irradiation fields are the standard thermal neutron irradiation mode, mixed 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 $0.1\ °C \cdot s^{-1}$.

**RESULTS:** Fig. 1 shows the Glow curves of $Al_2O_3$: 0.05 wt% Cr and $Al_2O_3$: 0.05 wt% Cr, 0.1wt% B for neutron and gamma mixed fields. The $Al_2O_3$: 0.05 wt% Cr and $Al_2O_3$: 0.05 wt% Cr, 0.1wt% B glow curves were the same; the glow peaks were located at 305 °C. In the TL sensitivity, $Al_2O_3$: 0.05 wt% Cr, 0.1wt% 0.1wt% B was lower than $Al_2O_3$:Cr. Although the B doping was not changed the trap state, it were generated lower the TL sensitivity. Fig. 2 shows the dose linearity for neutron and gamma mixed field of $Al_2O_3$: 0.05 wt% Cr and $Al_2O_3$: 0.05 wt% Cr, 0.1wt% B. Both TL phosphors were indicated the good linearity for irradiation time. Boron usage decreases sensitivity to gamma rays, but it is difficult to ascertain its sensitivity to neutrons. As a result, further studies will separate neutron and gamma ray.

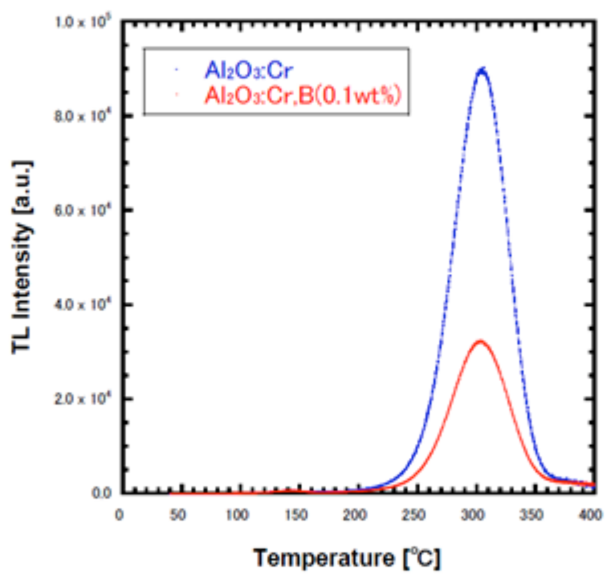


Fig. 1. Glow curves of $Al_2O_3$: 0.05 wt% Cr and $Al_2O_3$: 0.05 wt% Cr, 0.1wt% B for neutron and gamma mixed field..

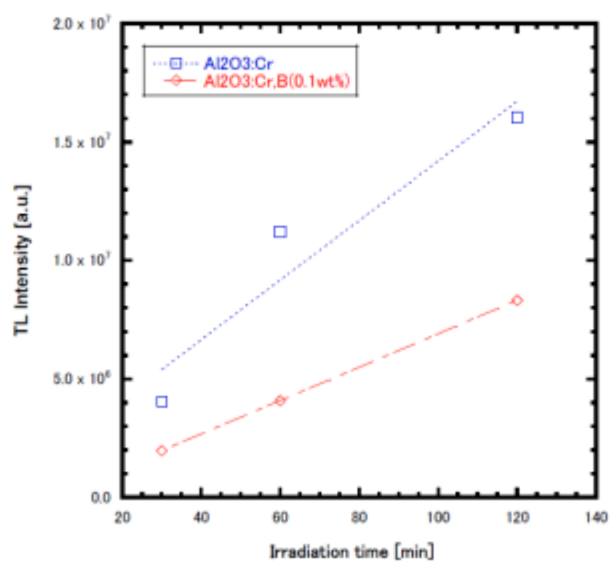


Fig. 2. Dose linearity for neutron and gamma mixed field of $Al_2O_3$: 0.05 wt% Cr and $Al_2O_3$: 0.05 wt% Cr, 0.1wt% B.

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[2] K. Shinsho *et al.*, Sensors and Materials., (2018) in press

## PR12-9 The Study for Development and Application of Tissue Equivalent Neutron Dosimeter

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**INTRODUCTION:** Recent years, the clinical application of boron neutron capture therapy (BNCT) has been started 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 spec-trum always spans a broad energy range, which results in different dose distribution and biological effects in tissue. Therefore, there are some difficulties of neutron dosimetry in clinical practice.

A radiochromic film (RCF) is one of the most useful devices for the QA of radiotherapy equipment. The advantages of RCFs are their high spatial resolution, small energy dependence, tissue equivalence, and self-development without processing in a darkroom[1]. A reflective-type RCF, e.g., GAFCHROMIC® EBT3, has been developed for qualitative dosimetry such as stereotactic irradiation (STI) and intensity modulated radiotherapy (IMRT)[2,3]. In this work, the authors investigated the response of reflective-type RCF for neutron beam as a tissue equivalent dosimeter.

**EXPERIMENTS:** A reflective-type RCF, GAFCHROMIC® EBT3 (Ashland Inc., Wayne, NJ, USA) using the KUR neutron source was evaluated in this study. The RCFs were handled by the recommendation outlined in the American Association of Physicists in Medicine Task Group No. 55 report[1].

For irradiation, 1.0 × 1.0 $cm^2$ pieces of the RCF were placed at the depth of 1 cm-12 cm from the water equivalent phantom surface at central axis and peripheral (Fig.1). Also, the film was placed beneath a gold foil for neutron dosimetry. The RCF pieces were irradiated by the neutron source, respectively.

**RESULTS:** Fig.1 shows the geometry of irradiation of RCFs by the KUR neutron source. Fig.2 shows the change of pixel values of RCFs irradiated by the source. There was significance difference of pixel values of irradiated RCFs by the neutron source. The change of pixel value of RCF piece between 1 cm and 12 cm at central axis was 11864, while the value with the peripheral was attenuated by 10807. Also, the value beneath the the gold foil was 23235. The central dose and peripheral dose converted by X-ray calibration of the RCF in the air were 237.2 cGy and 216.0 cGy, respectively. Besides neutrons, the RCFs were exposed to gamma rays coming from events taking place in the source itself and the surrounding moderator. Further analysis was needed of the response of RCFs with neutron spectrum and contributions of gamma rays using Monte Carlo simulation. However, the results suggested that the dosimetry using RCFs is important for a better knowledge of fast neutron flux distribution with the KUR neutron source. Moreover, it would be feasible for BNCT dosimetry in medical application.

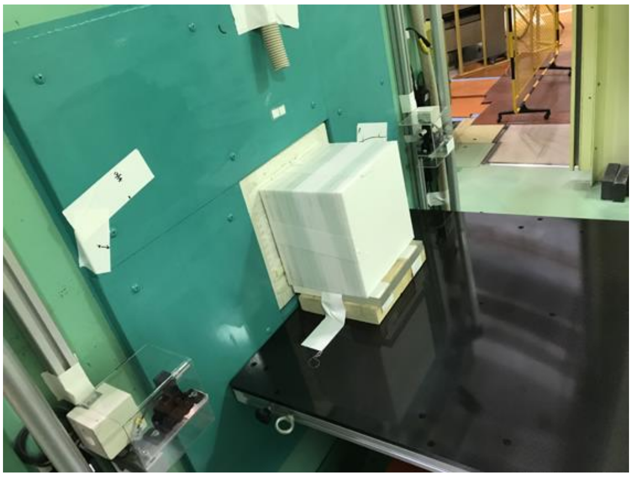

Fig. 1. The geometry of irradiation of RCFs by the KUR neutron source.

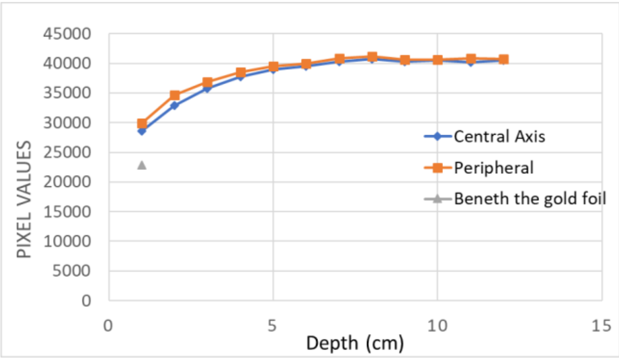


Fig. 2. The change of pixel values of the RCFs irradiated by the KUR neutron source.

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# PR12-10 Development and Evaluation of 3D Polymer Gel Dosimeter for the Measurement of dose Distribution in BNCT

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**INTRODUCTION:** Polymer gel dosimeters have been investigated for the three-dimensional (3D) dose measurement of the complex conformal dose distributions in the clinical applications [1]. These devices utilize radiation-induced polymerization reaction of vinyl monomer in the aqueous gel matrix to preserve information about the radiation dose. The 3D absorbed dose distribution is deduced from the created polymer distribution measured by imaging modalities such as MRI and Optical CT.

Polymer gel dosimeter is also regarded as tissue equivalent to neutron beam because the components are mainly water and a small amount of other chemicals consisting of carbon, nitrogen and oxygen. A further advantage of polymer gel dosimeters is that the interaction with neutron could be controlled by addition of some compounds with neutron-capture-nuclei such as $^{10}$B and $^{6}$Li. It means that each dose component might be distinguished from complex dose due to various primary and secondary radiations by the variety of elemental composition.

In this work, the influence of $^{10}$B and $^{6}$Li on the dose-response of polymer gel dosimeters was investigated by thermal neutron beam and compared the results of Monte Carlo simulations.

**EXPERIMENTS:** We prepared three types of polymer gel dosimeter in this work: (1) standard MAGAT polymer gel dosimeter composed of water (87 wt.%), methacrylic-acid (5 wt.%), gelatin (8 wt.%) and 2 mM of Tetrakis (hydroxymethyl) phosphonium chloride, (2) MAGAT added with 5 mM of $^{6}$Li enriched (96 at.% $^{6}$Li) lithium sulfate and (3) 25 mM boric acid of natural isotopic composition. The dosimeters prepared were subdivided into quartz test tubes and placed at 2.5, 5.0 and 7.5 cm depths in 10 cm × 10 cm × 10 cm cubic water phantom in order for the dosimeters to be irradiated with different neutron energy spectra.

The neutron irradiations were carried out using the epithermal neutron irradiation mode of Heavy Water Neutron Irradiation Facility at Kyoto University Reactor. The dosimeters were irradiated for 1 and 2 hours during 1 MW reactor operation at room temperature.

The read-out from the samples was performed using a 0.3 T MRI scanner (0.3 T AIRIS II Comfort, Hitachi medical) with a head coil the day after irradiation. A multiple spin-echo sequence was applied and the transverse relaxation rate $R_2$ ($=1/T_2$) was obtained as the function of absorbed dose.

Monte Carlo simulations were performed using PHITS code. Absorbed dose for the dosimeters were calculated by energy deposition of each secondary charged particle. LET distributions of the absorbed dose were also calculated using T-LET tally of PHITS code. The measured values using gold activation foils and BeO TLDs were utilized to normalize the calculated neutron fluence and gamma-ray dose respectively. The dose-response curves were obtained and the relationship between dose-response characteristics and LET distributions was analyzed.

**RESULTS:** The response of the three types of dosimeter to $^{60}$Co gamma-ray was almost identical. For neutron irradiations, the dose-response characteristics varied for different types of the dosimeter and neutron energy spectra. The relative efficiency of converting absorbed energy into the dosimeter response decreased monotonically up to one-fifth with increasing dose-mean LET. The more detailed analysis is underway.

(These results would be presented at 18$^{th}$ International Congress on Neutron Capture Therapy, 2018, ICNCT-18 in Taipei, Taiwan.)

**REFERENCE:**

[1] C. Baldock *et al.*, Phys. Med. Biol. **55** (2010) R1–63.

## PR12-11 Establishment of Beam-quality Estimation Method in BNCT Irradiation Field using Dual Phantom Technique

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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].

**METHODS:** One phantom was filled with pure water. The other phantom was filled with a water solution of lithium hydroxide (LiOH) capitalizing on the absorbing characteristics of lithium-6 (Li-6) for thermal neutrons.

Monte Carlo simulations were used to determine the ideal mixing ratio of Li-6 in LiOH solution. Changes in the depth dose distributions for each respective dose component along the central beam axis were used to assess the LiOH concentration at 0, 0.001, 0.01, 0.1, 1 and 10 weight percent.

Simulations were also performed with the phantom filled with 10 weight percent $^6$LiOH solution for 95%-enriched Li-6. A phantom was constructed containing 10 weight percent $^6$LiOH solution based on the simulation results.

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

**RESULTS:** Simulation results demonstrated that the absorbing effect for thermal neutrons occurred when the LiOH concentration was over 1%. The most effective Li-6 concentration was determined to be enriched $^6$LiOH with a solubility approaching its upper limit.

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 10%-$^6$LiOH phantom. It was confirmed that the dose contribution of fast neutrons is improved from approximately 10% in the pure water phantom, to approximately 50% in the 10%-$^6$LiOH phantom.

**CONCLUSION:** The dual phantom technique using the combination of a pure water phantom and a 10%-$^6$LiOH 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.

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

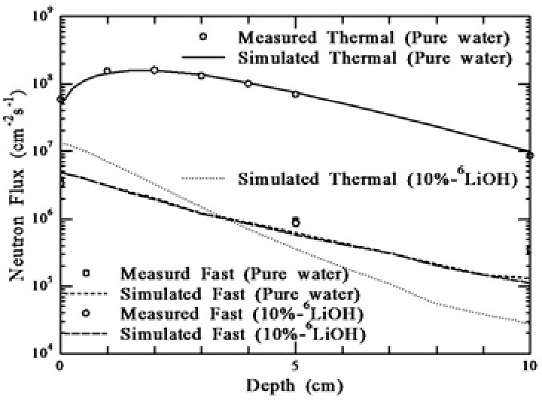


Fig. 1. Comparison between the measured and simulation results for the neutron-flux distributions in the pure water phantom and the 10%-$^6$LIOH phantom.

**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] Y. Sakurai *et al*., Med. Phys. **42** (2015) 6651-6657.

PR12-12

# Development of Real-time dose Monitor Using Prompt Rays Imaging Detector for Boron Neutron Capture Therapy

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**INTRODUCTION:** In order to improve the quality of BNCT treatment, it is desired to measure the boron concentration during treatment of BNCT[1]. The boron concentration in the blood has been evaluated using prompt gamma rays or ICP. In these evaluation methods, boron concentration can not be obtained during irradiation.

Therefore, we are developing a system that can detect boron concentration distribution in real time by detecting prompt gamma rays of 478 keV caused by nuclear reaction between boron and thermal neutrons. Since the annihilation gamma ray of 511 keV exists as the background of the BNCT irradiation field, it is necessary to discriminate between 478 keV and 511 keV. In order to discriminate between these two energies, the energy resolution needs to be 6.5% or less.

We constructed a prompt gamma-ray imaging detector system combining 8 × 8 arrayed $LaBr_3$(Ce) scintillator and 8 × 8 MPPC array. We report the outline and the result of the characteristic test.

**EXPERIMENTS:** The size of the $LaBr_3$ (Ce) scintillator is 50 mm x 50 mm x 10 mm and is divided into 8 x 8 arrays. The scintillator was set to an 8 x 8 array of MPPC.

The signal of 64 channels is amplified and shaped by the amplifier-shaper. Each analog signal was digitized by the ADC, and digital data was processed and stored by the PC. As a characteristic test, we tried to acquire the energy resolution and the two-dimensional distribution of 511 keV gamma ray using Na-22. In order to compare the performance of this system with arrayed scintillator, the same test was carried out for slab scintillators with the size of 50 mm x 50 mm x 10 mm.

**RESULTS:** For the arrayed scintillators, the energy resolution obtained from one pixel installed near the source was 7.6%, using gamma ray with Na-22 (Fig. 1). On the other hand, the energy resolution in each MPPC pixel of the non-arrayed scintillator was 6.6% on average.

One of the reasons that the energy resolution of arrayed scintillators is poor is that the peak at 511 keV is divided. The reason for the worse energy resolution is considered that the gain of each MPPC is not uniform. In the next step, we have a plan to set the different bias for each MPPC in order to improve energy resolution.

Fig. 2 shows the measurement results of two-dimensional incident position of incident gamma rays acquired by a not arrayed scintillator. It was confirmed that a two-dimensional distribution can be acquired by this system.

In the future, we will measure the boron concentration of boron samples using thermal neutron beam at KUR guide tube.

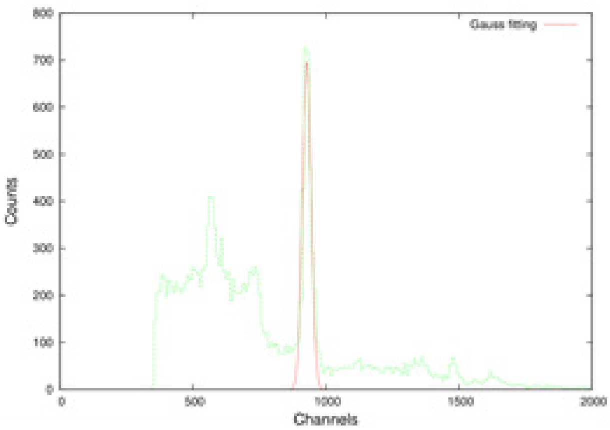


Fig. 1. The energy spectrum of gamma rays from Na-22.

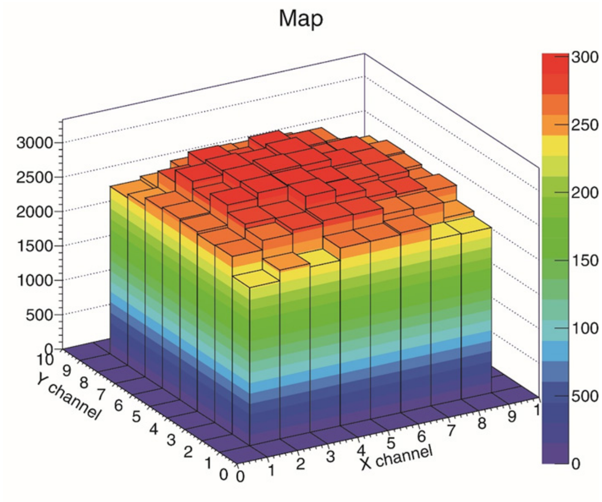


Fig. 2. Two-dimensional gamma rays image

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# PR12-13 Development of Novel Organic Scintillator for Fast Neutron and Its Evaluation in KUR

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**INTRODUCTION:** Neutron detection and imaging can find its application in various fields such as homeland security, crystallography, etc. [1, 2]. Up to now, $^3$He gaseous detectors have been used for neutron detection, because $^3$He has an unusually large cross-section for neutron capture (approximately 5300 barns for thermal neutrons) [3, 4]. However, the $^3$He sources are being depleted, and alternative suitable detection methods are required. $BF_3$ gas has been also used for a neutron detector. However, $BF_3$ gas has a big disadvantage since it is a toxic gas. On the other hand, neutron scintillator (solid state) is easy to handle to measure neutron.

To measure the neutron energy, Time-of-Flight (TOF) technique is required, and thus we need fast response scintillator. Some of halide scintillators show high light yield and fast decay time, however, most of them are hygroscopic. Therefore, we focused on the organic scintillator crystals, which showed fast decay time in the nanosecond range and no hygroscopic nature.

However, conventional organic scintillators have low melting temperatures and would degrade when overheated. Therefore, we have developed organic crystals for neutron scintillators with high melting temperatures and fast decay times.

In this research, we have developed novel organic scintillators, and test some of their characteristics such as the light output and radiation hardness.

**EXPERIMENTS:** In this paper, p-terphenyl and trans-stilbene (as reference), crystals were grown by self-seeding vertical Bridgman method using an enclosed chamber [5]. Raw material powder was charged into a double glass ampoule and the atmosphere in the chamber was replaced with high purity nitrogen. The ampoule was heated by a resistance heater and pulled down slowly at the rate of 0.03-0.06 mm/min. The grown crystals looked transparent. We tested the radiation hardness for our samples in Co-60 Gamma-ray Irradiation Facility of KUR.

**RESULTS:** We irradiated these samples four times with gamma rays from Co-60, and the total dose was around 5 Gy. After each irradiation, we obtained pulse height spectra for these samples irradiated with gamma rays from a Cs-137 source. We evaluated the light output from the Compton edge in the pulse height spectra. As shown in Fig. 1, we found light output of trans-stilbene was degraded after Co-60 irradiation as the dose increased. On the other hand, p-terphenyl seems to keep the light output even after 5 kGy (preliminary). In FY2018, also we would like to confirm the radiation harness for p-terphenyl and other novel organic sicnitillators.

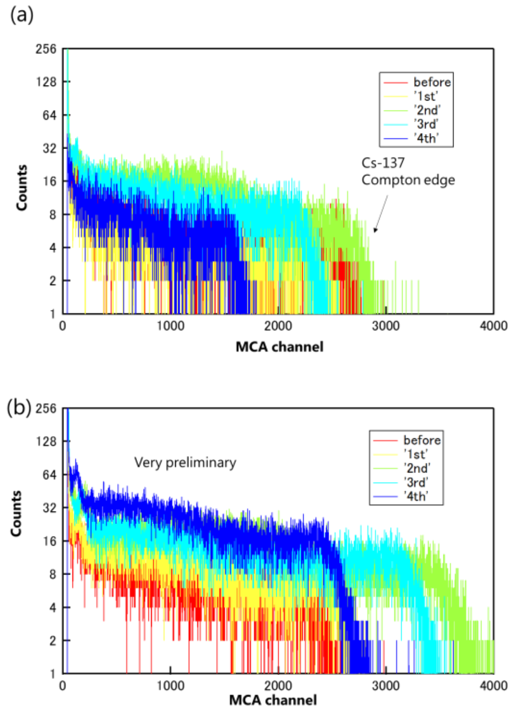


Fig. 1. Pulse height spectra of trans-stilbene (a) and p-terphenyl (b) measured under irradiation with gamma rays from Cs-137 before and after Co-60 irradiation.

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# PR12-14 Patient-Position Monitoring System for BNCT Irradiation

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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 preparing a patient-position monitoring system using a real-time range sensing devices. An outline of the monitoring system and initial test operation are described.

**MATERIALS AND METHODS:** An outline of the monitoring system is shown in Fig. 1. The monitoring system consists of sensor devices and an analyzer. Kinect sensor (MICROSOFT CORPORATION, USA) including a real-time range camera based on a time-of-flight method, was used to track a patient position. A range sensor of the Kinect has the following specifications: a frame rate of 30 fps and an image size of 512x424. Two Kinect sensors were placed at a distance of a few meter from a patient and viewing the patient from different directions. Data acquired with the each range camera, which is a 2D range image showing a pixel-by-pixel distance map to surface points of objects from the camera, was transferred to the position analyzer. In the analyzer, 3D surface data of a patient and surrounding objects was constructed from a set of range images, and then stored as a point cloud data represented in an arbitrarily specified Cartesian frame, such as a frame parallel and perpendicular to a beam axis.

These data can be used for patient position monitoring not only by visual observation but also by quantitative evaluation. Also, displacement due to a set-up error and motion can be evaluated by comparing a position determined in a treatment planning system with a position measured by this system.

**RESULTS:** An initial test operation of the Kinect sensor was conducted. The sensor was placed at a distance of 200 cm from a reference object. It was confirmed that the indication of the distance was correctly displayed. However, the observed distance showed somewhat large time fluctuation in a range of ±30 mm, which was considered as random noise. A moving averaging method was applied to reduce the fluctuation, and it was confirmed that the average over 100 frame could reduce the fluctuation to < 5 mm. Also, it was confirmed that the analyzer could calculate 3D surface data from a set of range data and display a range image with a viewing axis parallel to a beam axis.

**CONCLUSION:** An outline of the monitoring system and initial test operation results were described. It was confirmed that the range sensing devices used in the system have sufficient accuracy and the position analysis can be correctly performed. The system can be expected to work effectively for monitoring a patient position. Functions to calculate a displacement are needed to evaluate a set-up error and motion, and are currently under development

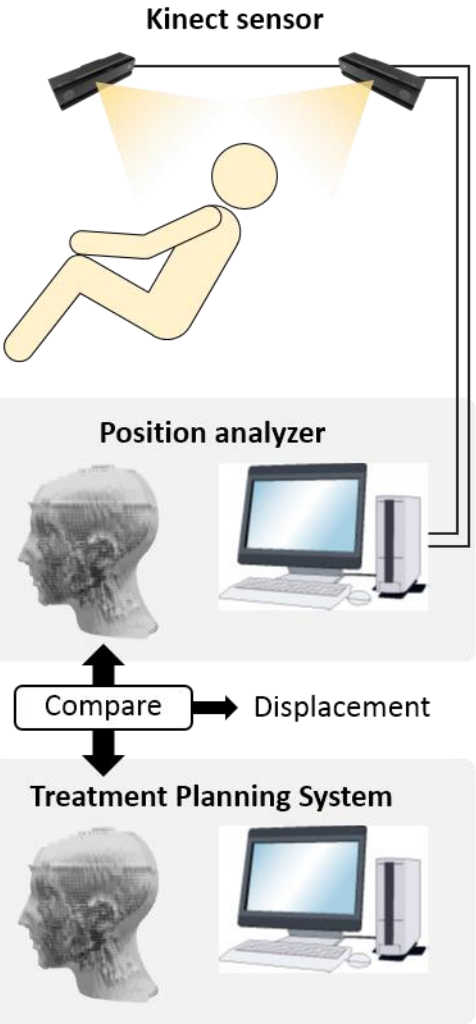


Fig. 1. An outline of the patient position monitoring system.