I-1. PROJECT RESEARCHES

Project 9

PR9 Advancement of integrated system for dose estimation in BNCT

Y. Sakurai

Institute for Integrated Radiation and Nuclear Science, Kyoto University

BACKGROUNDS AND PURPOSES:

There are a number of subjects, which should be improved for the further advance and generalization of boron neutron capture therapy (BNCT). In the viewpoints of medical physics and engineering, the advance for dose estimation is one of the important subjects. The purposes of this project research are the advance for various dose estimation methods, and the establishment of an integrated system for dose estimation in BNCT.

In the first year of this research project, 2020, the advancement for the respective dose estimation methods were forwarded mainly using Heavy Water Neutron Irradiation Facility (HWNIF) and E-3 Neutron Guide Tube (E-3) at KUR. In addition, the integrated system was considered for the simultaneous usage of several dose estimation methods.

RESEARCH SUBJECTS:

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

ARS-1 (R2P9-1): Establishment of characterization estimation method in BNCT irradiation field using Bonner sphere and ionization chamber (IV). (Y. Sakurai, S. Shiraishi, N. Matsubayashi, A. Sasaki, Y. Kakimoto, T. Naito, H. Matsunaga, T. Nakamura, R. Narita, M. Nojiri, H. Kato, T. Takata, H. Tanaka)

ARS-2 (R2P9-2): Study on new type of neutron spectrometer for BNCT. (K. Watanabe, A. Ishikawa, A. Uritani, S. Yoshihashi, A. Yamazaki, Y. Sakurai)

ARS-3 (R2P9-3): Response characteristic measurements of lithium-glass scintillators used in the BSS for the intense neutron beams. (A. Masuda, T. Matsumoto, H. Tanaka, D. Ito, H. Harano, Y. Sakurai, T. Takata)

ARS-5 (R2P9-5): Neutron measurement by using the self-activation of iodine-added liquid scintillators. (A. Nohtomi, Y. Kanzaki, G. Wakabayashi, Y. Sakurai, T. Takata)

ARS-6 (R2P9-6): Development of absolute epi-thermal and fast neutron flux intensity detectors for BNCT (I. Murata, K. Aoki, Y. Miyaji, D. Hatano, S. Kusaka, H. Tanaka, Y. Sakurai, T. Takada)

ARS-8 (R2P9-8): Study for microdosimetry using silicon-on-insulator microdosimeter in the BNCT irradiation field (IV). (Y. Sakurai, N. Ko, T. Takata, H. Tanaka, T. L. Tran, J. Davis, S. Guatelli, A. Rozenfeld, N. Kondo, M. Suzuki)

ARS-9 (R2P9-9): Measurement of thermal neutrons and gamma rays in BNCT beam with polymer gel detector. (K. Tanaka, Y. Sakurai, T. Kajimoto, A. Mitsuyasu, S. Hayashi, Hiroki Tanaka, T. Takata, G. Bengua, S. Endo)

ARS-10 (R2P9-10): Development of two dimensional neutron detector in BNCT irradiation field. (H. Tanaka, T. Takata, Y. Sakurai, S. Uno)

ARS-11 (R2P9-11): Study of neutron fluence and gamma ray distribution measuring using thermoluminescence slabs. (K. Shinsho, R. Oh, M. Tanaka, S. Yanagisawa, H. Tanaka, T. Takata1, G. Wakabayashi, Y. Koba)

ARS-13 (R2P9-13): Development and evaluation of 3D gel dosimeter for the measurement of dose distribution in BNCT. (S. Hayashi, Y. Kakimoto, T. Takata, M. Suzuki, Y. Sakurai)

ARS-14 (R2P9-14): Establishment of beam-quality estimation method in BNCT irradiation field using dual phantom technique (IV). (Y. Sakurai, N. Kondo, T. Takata, H. Tanaka, M. Suzuki)

ARS-15 (R2P9-15): Development of a prompt gamma-ray imaging detector using an 8 x 8 arrayed $LaBr_3$(Ce) scintillator for BNCT. (K. Okazaki, T. Takata, Y. Sakurai, H. Tanaka)

ARS-16 (R2P9-16): Development of fiber-reading radiation monitoring system with an optical fiber and red-emitting scintillator at the ^{60}Co radiation facility. (S. Kurosawa, H. Tanaka, S. Kodama, Y. Kurashima, A. Yamaji)

ARS-17 (R2P9-17): 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. (S. Komura, T. Mizumoto, Y. Sakurai, T. Takata, T. Tanimori, A. Takada)

ARS-18 (R2P9-18): Feasibility study for quality assurance and control for irradiation field in BNCT. (S. Nakamura, M. Takemori, K. Iijima, S. Imamichi, H. Nakayama, T. Chiba, M. Masutani, Y. Sakurai, H. Tanaka, T. Takata, M. Suzuki, H. Igaki, H. Okamoto)

ARS-19 (R2P9-19): Evaluation of thermal neutron irradiation field for semiconductor device irradiation. (H. Tanaka, T. Takata, N. Matsubayashi, Y. Sakurai)

ARS-20 (R2P9-20): Optimization of bolus shape for boron neutron capture therapy - examination using simple shaped phantom for experimental verification -. (T. Takata, H. Tanaka, A. Sasaki1, Y. Sakurai, M. Suzuki)

ARS-23 (R2P9-23): Verification of the annealing capability of boric acid-infused PVA-GTA-I gel dosimeter. (H. Yasuda, JE. Taño, CAB. Gonzales, Y. Sakurai)

ARS-24 (R2P9-24): Establishment of quantitative measurement of boron concentration distribution in vivo by imaging of prompt gamma rays. (S. Komura, T. Mizumoto, T. Mitani, Y. Sakurai, N. Kondo, M. Suzuki, T. Takata, T. Tanimori, A. Takada)

ARS-4, ARS-12, ARS-21 and ARS-22 could not be performed because of the influence of COVID-19 infection. For ARS-7, the submission of the report was suspended because of the reason for the patent application. So, the reports for these subjects are not appeared.

PR9-1 Establishment of characterization estimation method in BNCT irradiation field using Bonner sphere and ionization chamber (IV)

Y. Sakurai, S. Shiraishi[1], N. Matsubayashi[1], A. Sasaki[1], Y. Kakimoto[1], T. Naito[1], H. Matsunaga[1], T. Nakamura[1], R. Narita[1], M. Nojiri[1], H. Kato[1], T. Takata and H. Tanaka

Institute for Integrated Radiation and Nuclear Science, Kyoto University
[1]Graduate School of Engineering, Kyoto University

INTRODUCTION: Development in accelerator-based irradiation systems for BNCT is underway. In the near future, BNCT using these newly developed 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 QA/QC system for BNCT.

As part of the QA/QC system, we are developing estimation method for neutron energy spectrum using Bonner sphere [1]. 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 detector, etc.

MATERIALS AND METHODS: In the neutron energy spectrometry by Bonner-sphere, 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) [2].

On the assumption of the application in the standard epi-thermal neutron irradiation mode of Heavy Water Neutron Irradiation Facility (HWNIF), the combination of the moderators for boron-10 concentration and diameter was optimized by HIS. Based on this optimization, a prototype of RBS was made. Some experiments were performed for the characteristic verification of the prototype RBS at HWNIF.

RESULTS: The configuration of the RBS was decided as follows. A five-layer concentric spherical acrylic shell is used as a container. Each acrylic wall is 1 mm in thickness. The moderator injection part is 9 mm in thickness for each layer. Pure water and 0.14-wt% boric acid water for boron-10 were used as liquid moderators. LiCaF scintillation detector and gold wire were used as the detectors. Figure 1 shows the combination of moderator injection patterns for RBS, used in the experiments.

In both cases of LiCaF scintillation detector and gold wire, the measured values were larger than the calculated values. The concentration of boric acid was assumed to be 0.14 wt% in the calculation, but the actual concentration was 0.12 wt%. This is considered to be one of the reasons for the disagreement. The other possible reasons are the differences in sizes between the actually prepared spectrometer, field size, beam directionality, etc..

Figure 2 shows the comparison between the nominal spectrum and unfolded spectrum. Unfolding was performed by GRAVEL using the response function of each Bonner sphere corrected by multiplying the ratio for measured/calculated values. The nominal spectrum of the epi-thermal neutron irradiation mode was input as an initial guess. The spectrum obtained by the unfolding reproduced the nominal spectrum relatively well, but the absolute value was overestimated about two times.

CONCLUSION: We have plan to perform (1) simulation with boric acid concentration set to 0.12 wt%, and (2) preparation of a Bonner sphere spectrometer including the remote mechanism for the supply and drainage of the liquid moderators.

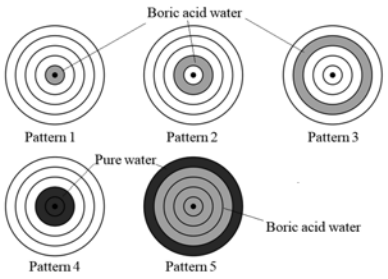

Fig. 1. The used combination for RBS.

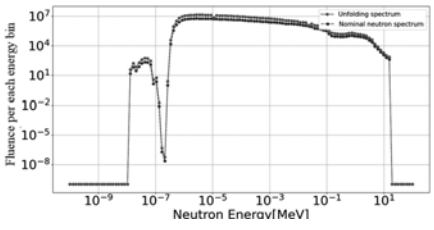

Fig. 2. Comparison between the nominal spectrum and the unfolded spectrum.

REFERENCES:

[1] S. Shiraishi *et al.*, Appl. Radiat. Isot. **163** (2020) 109213.
[2] H. Ueda, Doctoral Thesis (2016).

PR9-2

Study on New Type of Neutron Spctrometer for BNCT

K. Watababe, A. Ishikawa, A. Uritani, S. Yoshihashi, A. Yamazaki and Y. Sakurai[1]

Graduate School of Engineering, Nagoya University
[1] Institute for Integrated Radiation and Nuclear Science, Kyoto University

INTRODUCTION: Boron neutron capture therapy (BNCT) is a combined modality of radiotherapy and chemotherapy for cancer treatment. In the BNCT, a boron-containing agent is injected in patient's blood and concentrated into tumor cells. And then, the patient is irradiated with neutrons externally and $^{10}B(n,a)$ reactions are induced in tumor cells. Recently, an accelerator-driven neutron source has actively been developed instead of nuclear reactors, owing to its simplicity of management. In commissioning phase of the facilities, characterization of the irradiation field, such as neutron intensity, the neutron energy spectrum and gamma-ray components, should be characterized in order to assure designed ones.

We are developing a new neutron energy spectrometer using the optical fiber type neutron detector. The conventional technique for neutron spectrometry is the Bonner sphere method. Generally, energy response in epithermal region in moderation based neutron spectrometers is not sufficient. In order to improve the energy response, the number of detectors with different responses should be increased. In our proposed system, the thermal neutron flux profile in a liquid moderator is measured by the optical fiber type neutron detector. So far, in order to realize the optical fiber type neutron detector showing a neutron peak in the pulse height spectrum, bright neutron scintillators, such as Eu:$LiCaAlF_6$ or LiF/Eu:CaF_2 eutectics, have been used [1]. Recently, we attempted to replace them with the faster Li glass scintillator. Using an optical fiber with high numerical aperture, we successfully demonstrated the optical fiber type neutron detector showing a neutron peak by using the Li glass scintillator [2]. In this study, we evaluated gamma-ray sensitivity of the optical fiber type neutron detector using the Li glass scintillator.

EXPERIMENTS: We fabricated the optical fiber type neutron detectors using Li glass scintillator. A small piece of the Li glass scintillator was mounted at a tip of the optical fiber. The scintillator was coated with a reflector powder and then covered with a heat shrinking tube for light shielding.

First of all, we evaluated the neutron response at the E3 port in the Kyoto University Reactor. In this experiment, signal pulse height spectra for $^6Li(n,t)$ reactions were obtained. And then, the same detectors were irradiated with gamma-rays at Co-60 gamma-ray irradiation facility in Nagoya University to evaluate the gamma-ray sensitivity. Figure 1 shows photographs of thermal neutron and Co-60 gamma-ray irradiation experiments.

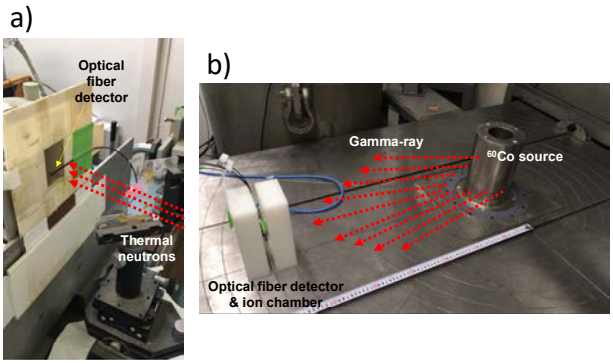

Fig. 1 Experimental setup. a) E3 port in KUR for thermal neutron irradiation. b) Co-60 gamma-ray irradiation facility in Nagoya University.

RESULTS: Figure 2 shows signal pulse height spectra obtained when the optical fiber type neutron detector using a Li glass scintillator of 18 μg was irradiated with thermal neutrons and Co-60 gamma-rays. The fabricated optical fiber type neutron detector shows a clear neutron peak. When irradiating with Co-60 gamma-rays, signals with slightly lower pulse heights compared with a neutron peak were recorded. In order t define the gamma-ray sensitivity of the detector, we have to determine the discrimination level for neutron events. When setting the discrimination level at a valley in the pulse height spectrum, the gamma-ray sensitivity was evaluated to be 4 cps at 1.9 Gy/h irradiation, which is higher dose rate compared with expected BNCT irradiation fields. This gamma-ray sensitivity is sufficiently low because the counting rate of the detector in typical BNCT irradiation field is expected to be more than 10^3 cps.

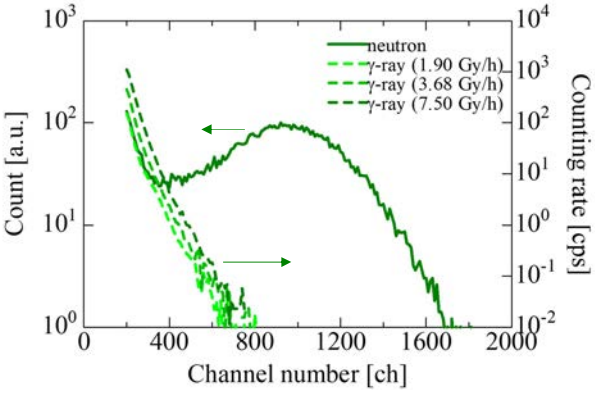

Fig. 2 Signal pulse height spectra obtained when the optical fiber type neutron detector using a Li glass scintillator of 18 μg was irradiated with thermal neutrons and Co-60 gamma-rays.

REFERENCES:

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[2] A. Ishikawa *et al.*, Sensors and Materials, **32**, 1489-1495 (2020).

PR9-3 Response characteristic measurements of lithium-glass scintillators used in the BSS for the intense neutron beams

A. Masuda, T. Matsumoto, H. Tanaka[1], D. Ito[1], H. Harano, Y. Sakurai[1] and T. Takata[1]

National Metrology Institute of Japan, National Institute of Advanced Industrial Science and Technology
[1]Institute for Integrated Radiation and Nuclear Science, Kyoto University

INTRODUCTION: Characteristics of the intense neutron beam should be evaluated to ensure the performance and safety of boron neutron capture therapy (BNCT). The Bonner unfolding method is commonly used when the time-of-flight method is not suitable for the moderated neutron source. Although the Bonner unfolding method has been successfully applied to measure weak beams at a BNCT facility[1], direct measurements of the therapy-level intense neutron beams are required. A new Bonner sphere spectrometer (BSS) using a pair of lithium-glass scintillators coupled with current-integration-operated PMTs has been developed to measure the BNCT beams[2]. Characteristics of the lithium-glass scintillators should be evaluated in detail for the practical use.

EXPERIMENTS: Directional-dependent response characteristics of the pair of lithium-glass scintillators are focused in this study. The well-collimated thermal neutron beam of the B-4 beam hole of the KUR[3] was used to evaluate the directional dependency precisely. The pair of the lithium-glass scintillators of GS20 (95% ^{6}Li, 10 mm in diameter and 2 mm in thickness) and GS30 (99.99% ^{7}Li, 10 mm in diameter and 2 mm in thickness) was set and irradiated by the thermal neutron beams from the four different directions as shown in Fig. 1. The effect of the γ-ray induced signal was corrected by the difference between the GS20 and GS30[4]. The same measurements using the standard mixed neutron irradiation mode were also carried out in the heavy water irradiation facility of the KUR[5] to simulate the actual therapy environment.

RESULTS: Relative efficiency to the direction (1) is shown in Fig. 2. Output variation of the reactor for each measurement was corrected using a fission chamber. A significant directional dependency was observed for the well-collimated thermal neutron beam. Decreasing of efficiency in the side irradiations is considered to be due to the effect of the absorption by the scintillator. Difference between the (2) and (3) can be explainable by effects on the subtraction of γ-ray induced signal. On contrast, the directional dependency was not so much observed in the heavy water irradiation facility. Since the irradiation room is not large, room-scattered low-energy neutrons seems to be dominant for the detector.

The results have shown that the efficiency of the detector has strong directional dependency and it should be considered when it is used in the actual treatment rooms of BNCT. Although the directional dependency can be negligible when used in the large moderator spheres of the BSS, it might be problematic when used with small spheres or without spheres.

The characteristics of the detector will be investigated more, and it will be utilized to improve the measurements and analysis of the BSS measurement in BNCT facilities.

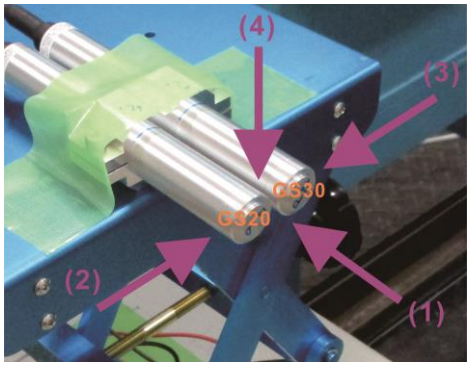

Fig. 1. The pair of GS20 and GS30 scintillators irradiated by the for directions: (1) top, (2) GS20 side, (3) GS30 side and (4) equivalently side.

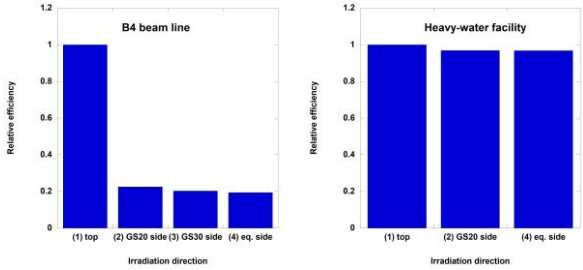

Fig. 2. Directional dependency of output from the pair of the lithium-glass scintillators.

REFERENCES:

[1] A. Masuda et al., Appl. Radiat. Isot., 127 (2017) 47-51.
[2] A. Masuda et al., 2018 IEEE NSS/MIC Proceedings (2019) 10.1109/NSSMIC.2018.8824697.
[3] Y. Saito et al., Nucl. Instr. Meth. Phys. Res. A, 651 (2011).
[4] T. Matsumoto et al., Radiat. Prot. Dosim., 188 (2019) 117-122.
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This study is supported by Grant-in-Aid for Scientific Research, Japan Society for the Promotion of Science (JSPS KAKENHI Grant Number 19K12638).

PR9-4 Neutron measurement by using the self-activation of iodine-added liquid scintillators

A. Nohtomi, Y. Kanzaki, G.Wakabayashi[1], Y.Sakurai[2] and T.Takata[2]

Graduate School of Medicine, Kyushu University
[1] Atomic Energy Research Institute, Kindai University
[2] Institute for Integrated Radiation and Nuclear Science, Kyoto University

INTRODUCTION: In our previous report, applicability of iodine-added plastic scintillator was studied at rather intense neutron field like BNCT at one of KUR irradiation facility [1]. In the study, thermal neutron was successfully detected by using the self-activation of iodine which was added to plastic scintillators. As with iodine-containing inorganic scintillators, NaI and CsI [2,3], the method is highly sensitive even for short time irradiation. On the other hand, the tuning of sensitivity is possible for this method. In addition, another benefit of this method is no production of byproduct activity, ^{24}Na and ^{134m}Cs. In the present work, applicability of iodine-added liquid scintillator was studied for comparison with the iodine-added plastic scintillator.

EXPERIMENTS: Iodine-added liquid scintillator was prepared by adding iodobenzen (C_6H_5I) to commercially available liquid scintillator (Insta-Gel Plus, Perkin Elmer). 20 ml polyethylene vials were used as the container of liquid scintillator. The surface of polyethylene vial was covered with Teflon tape as scintillation reflector (Fig.1). The concentration of iodobenzene in liquid scintillator was adjusted to be 1 wt% and 0.1 wt%. Two liquid scintillators were irradiated at Rail Device of the Heavy Water Neutron Irradiation Facility with OO-0000F mode (1MW) [4]. Those scintillator vials were put at the Bismuth Surface (thermal neutron flux : ~10^9 n/cm^2/s) during 10 seconds for 1 wt% scintillator and 100 seconds for 0.1 wt% one. After the termination of each irradiation, the output signal of scintillators was read out by PMT and MCA every one minute continuously.

Fig. 1. Appearance of iodine-added liquid scintillator.

RESULTS: Figure 2 shows decay curves of count rate observed by the iodine-added liquid scintillators after the termination of irradiation. In addition to ^{128}I decaying components with half-life of 25 min, small amount of unknown decaying components with half-life of 900 min are seen for both 1 wt% and 0.1 wt% scintillators. The count rate for 1 wt% at the termination of irradiation is 1.09 x 10^6 cpm and that for 0.1 wt% is 1.02 x 10^6 cpm.

$$1\text{wt\%}: 1.09\times10^6\times(1/2)^{\frac{x}{25}}+270\times(1/2)^{\frac{x}{900}}+230$$

$$0.1\text{wt\%}: 1.02\times10^6\times(1/2)^{\frac{x}{25}}+3000\times(1/2)^{\frac{x}{900}}+300$$

Because the irradiation time of neutrons is very shorter than the half life of ^{128}I (25 min), total amount of ^{128}I activity generation is approximately proportional to the product of the neutron irradiation time and the total number of ^{127}I atom existing in the scintillators. Therefore, it is consistent that the initial count rate (cpm) for 1 wt% scintillator after 10 seconds irradiation is almost equivalent to that for 0.1 wt% scintillator after 100 seconds irradiation (~1 x 10^6 cpm).

We could confirm that the unknown component was ^{24}Na by the gamma-ray spectroscopy using HPGe. It has been found that such small amount of contamination of ^{23}Na in liquid scintillator is not desirable for the application of iodine-added liquid scintillator to intense BNCT neutron field.

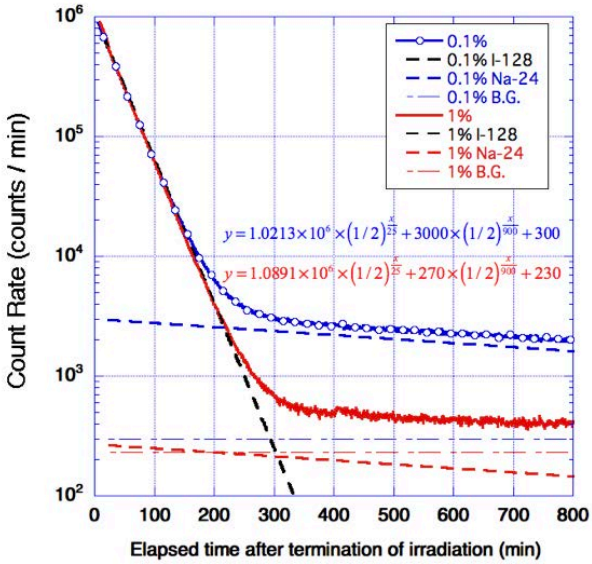

Fig. 2. Decay curves of count rate observed by the iodine-added liquid scintillators.

ACKNOWLEDGEMENT: This work was supported by JSPS KAKENHI Grant Number JP19K08202.
Mr. N. Sakamoto and Mr. H. Maeda of Kyushu University are deeply appreciated for their preparation of iodine-added liquid scintillators.

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[1] A. Nohtomi *et al.*, KURRI progress report 2019.
[2] G. Wakabsyashi *et al.*, Radiol. Phys. Technol., **8** (2015) 125-134.
[3] A. Nohtomi *et al.*, JPS Conf. Proc., **24** (2019) 011041.
[4] Y. Sakurai and T. Kobayashi, Nucl. Instr. and Meth., **A453** (2000) 569-5.

PR9-5

Development of Absolute Epi-thermal and Fast Neutron Flux Intensity Detectors for BNCT

I. Murata, K. Aoki, Y. Miyaji, D. Hatano, S. Kusaka, H. Tanaka[1], Y. Sakurai[1], T. Takada[1]

Graduate School of Engineering, Osaka University
[1]Institute for Integrated Radiation and Nuclear Science, Kyoto University

INTRODUCTION: BNCT is a promising cancer therapy which can kill tumor cells while suppressing exposure dose to normal tissues. Normally, the neutron field of BNCT, which is produced by a nuclear reactor or an accelerator, has an energy distribution spreading within thermal, epi-thermal and fast neutron regions. Because epi-thermal neutrons are generally used for BNCT, we must measure the epi-thermal neutron flux intensity to evaluate the therapeutic effect and patient's exposure dose. In addition, we also have to evaluate the exposure dose of the fast neutrons that may be harmful to the human body. However, it is quite difficult to know such intensities directly and accurately, because there is no suitable neutron spectrometer and no activation material covering epi-thermal or fast neutrons separately. The objective of this work is hence to develop new detectors to precisely measure the absolute integral flux intensities of epi-thermal (0.5 eV ~ 10 keV) and fast neutrons (10 keV ~ 1 MeV).

EXPERIMENTAL METHODS: Epi-thermal neutron detector was designed to reduce sensitivities to thermal and fast neutrons by using cadmium and polyethylene. Fast neutron detector reduces sensitivities to thermal and epi-thermal neutrons by using cadmium, B_4C and polyethylene. We performed numerical simulations by MCNP5 and $^{71}Ga(n, \gamma)^{72}Ga$ reaction was determined to use as an activation reaction. In order to test the performance of the epi-thermal and fast neutron detectors, verification experiments were conducted at KUR, Kyoto University.

RESULTS and DISCUSSION: Figure 1 shows the design result of the epi-thermal neutron detector. Figure 2 and 3 are sensitivities of epi-thermal and fast neutron detectors, respectively. Epi-thermal neutron detector still has a small but non-negligible sensitive to fast neutrons as

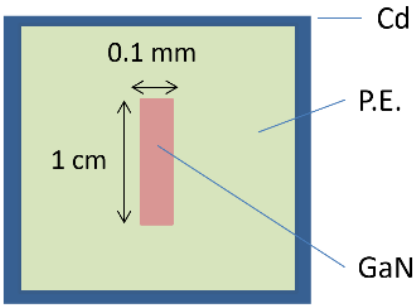

Fig. 1. Calculation model of epi-thermal neutron detector.

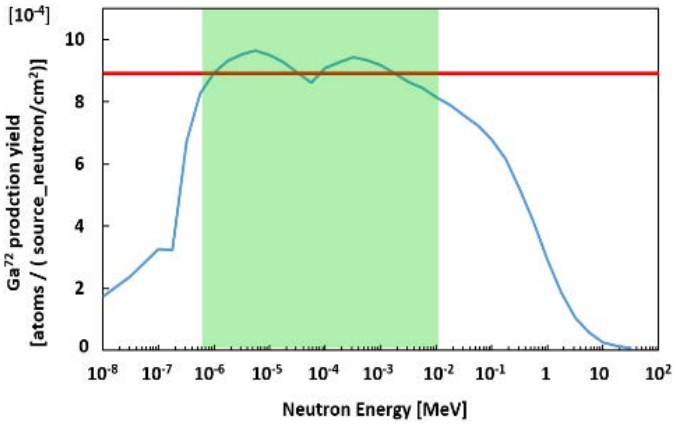

Fig. 2. Sensitivity of epi-thermal neutron detector.

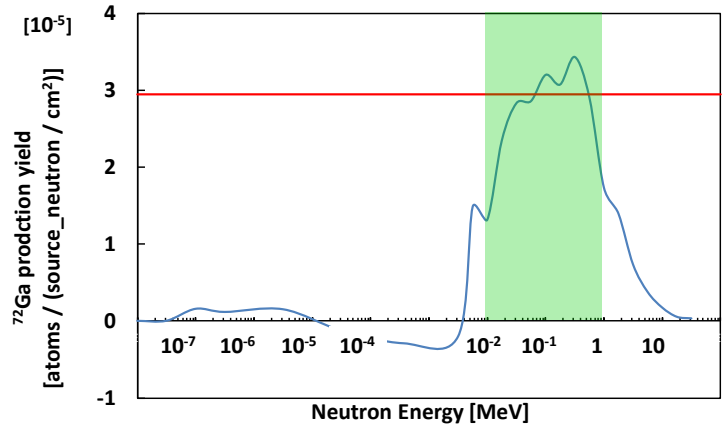

Fig. 3. Sensitivity of fast neutron detector.

in Fig. 2. The experimental results thus show a slight overestimation in the epi-thermal neutron flux intensity, i.e., C/E = 0.61. Similarly, since fast neutron detector has a little sensitivity to thermal and epi-thermal neutron, a considerable overestimation, C/E = 0.28, was observed. This is mainly because the real spectrum has a peak in the epi-thermal energy region.

PR9-6 Characterization of Active Neutron Detector for Boron Neutron Capture Therapy

M. Takada, S. Endo[1], H. Tanaka[2], T. Nunomiya[3], K. Aoyama[3], T. Nakamura[3]

Department of Applied Physics, National Defense Academy of Japan
[1]Graduate School of Engineering, Hiroshima University
[2]Institute for Integrated Radiation and Nuclear Science, Kyoto University
[3]Fuji Electric Co. Ltd.

INTRODUCTION: Boron Neutron Capture Therapy (BNCT) is a binary radiotherapy method developed to treat patients with certain malignant tumors. BNCT has been widely used at nuclear reactors. Currently, several accelerator-based BNCT facilities are being developed and clinical trials are conducted. Intensity of neutron beam produced from the neutron accelerator could be varied with time due to instability of the accelerator condition and the neutron target degression. Now, thermal neutron flux is measured before and after the neutron irradiation using offline gold activation method. However, real-time neutron beam measurement is required during the neutron irradiation.

The active neutron detector, which composes of thin silicon sensor and ultra thin LiF neutron converter with around 0.1μm thickness, measures the BNCT neutron beam over $1x10^9$ (cm^{-2} s^{-1}), separately from high dose-rate gamma rays around 500 mGy/h [1-3]. The detection efficiency of the real-time neutron detector for thermal neutron was experimentally obtained. Neutron responses for a wide range of neutron converter thickness have been measured and simulated. Neutron detectors have been developed to measure epithermal and fast neutrons produced from the neutron target.

These neutron detectors will be applied to monitor the target neutrons during the BNCT treatments by installing the neutron detector near the neutron target. There, photon dose rates are dramatically increased than that at the neutron irradiation position. It is difficult to separate neutron events from photons under high-dose-rate photon fields because of interruption by pile-up signals. A neutron detector was tried to improve the separation property of neutron and photon events by acquiring larger particle energies of summing both triton and alpha partilces produced by the $^6Li(n,t)^4He$ reaction.

EXPERIMENTS: The neutron measurements were performed at the heavy water irradiation facility of research reactor of KURNS. Two thin silicon sensors were faced each other and an ultra thin LiF neutron converter, evaporated on a thin Mylar film, is inserted between them. Summed both signals from the two Si sensors were fed to a pre-amplifier outside of the irradiation room, and then, acquired using fast digital pulse-shaping processor. The thermal neutron beam fluxes and cadmium ratios are changed by opening a Cd aperture slit at an upward of the irradiation location. Channel number of multi-channel analyzer was converted to particle energies by measuring alpha sources at our university before the neutron irradiation.

RESULTS: Figure.1 shows the pulse heights of thermal neutrons measured using the Si neutron detector, normalized with the neutron fluences. Here, the neutron fluences are referred from the previous evaluation values. Three main peaks are observed in the plot, (A)-(C). The peak (A) is created by detecting both alpha particles and tritons. Tritons are transmitted through the thin film and loss their energies, however, alpha particles loss no energy in the film. The peak (B) is created by detecting alpha particles without any energy loss and tritons transmitted through the film. Neutron peak is split due to different particle energy losses in the film. These split peaks are resolved by evaporating the LiF on the Si sensor, directly and removing the film. Unfortunately, the creation of peak (C) has never been understood. Events (D) and (E) are observed due to detection of pile-up signals and mainly electrons produced by gamma rays, respectively.

During the neutron irradiation, the neutron peaks show constant particle energy and no peak shift was observed. Until now, neutron damage on Si sensor has never been measured. Up to $6.5x10^8$ (cm^{-2} s^{-1}) neutron flux, linearity of count rates was obtained. BNCT neutron beam was measured directly, using the compact neutron detector. We will measure thermal neutron fluxes under high photon-dose-rates environments.

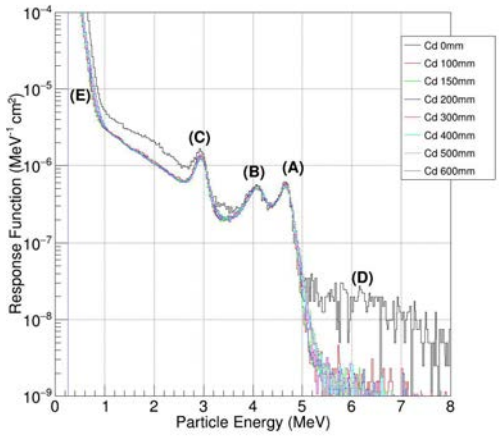

Figure 1. Pulse heights of neutron detector, which is consisted of thin Si sensor and ultra-thin LiF neutron converter, measuring BNCT neutron beam. Thermal neutron fluxes were changed by opening the Cd aperture window.

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PR9-7 Study for microdosimetry using silicon-on-insulator microdosimeter in the BNCT irradiation field (IV)

Y. Sakurai, N. Ko[1], T. Takata, H. Tanaka, T. L. Tran[1], J. Davis[1], S. Guatelli[1], A. Rozenfeld[1], N. Kondo and M. Suzuki

Institute for Integrated Radiation and Nuclear Science, Kyoto University
[1]Kansai BNCT Medical Center, Osaka Medical College
[2]Centre for Medical Radiation Physics, University of Wollongong

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 2020, a new-type silicon microdosimeter and its application for boron neutron capture therapy (BNCT) were investigated by Monte Carlo simulation.
MATERIALS AND METHODS: Two detector configurations were investigated, based on the current 3D mushroom microdosimeter. The first structure consists of a cylindrical p+ core electrode through the center of the SV with n+ ring electrode wrapped around the outside of the SV. The second structure consists of a cylindrical n+ core electrode through the center of the SV with p+ ring electrode wrapped around the outside of the SV. Each SV has a diameter and height of 10 μm and the pitch between each individual SV is 40 μm to reduce cross talk between neighboring row of detectors. A total of 2500 individual SVs were connected in an array with odd and even detector row readout channels.
PHITS was used for this study. The T-deposit tally, which scores dose and event-by-event deposition energy distribution was used to calculate the energy deposited inside the SV of the mushroom microdosimeter. The microdosimetric spectrum (frequency mean and dose mean lineal energy distribution) were calculated by dividing the deposited energy by the average chord length of the SV.
The neutron response of the new-type detector were investigated using the neutron source for the mixed neutron irradiation mode of Heavy Water Neutron Irradiation Facility installed in Kyoto University Reactor (KUR-HWNIF) [4].
RESULTS: The ion track of 1.47 MeV alpha particles produced inside the p+ region is shown in Figure 1. It was found the total flux inside the SV was higher for the p+ core through the center. This was due to the fact some of the alpha particles escaped from the p+ ring.
The number of alpha particles depositing it's full energy (1.47 MeV) was also found to be higher with the p+ core design, shown in Figure 2. The range of 1.47 MeV alpha particles inside silicon is approximately 5 μm. Most of the alpha particles generated inside the p+ core will deposit all its energy inside the SV. However, the alpha particles generated inside the p+ ring will only deposit a portion of its energy inside the SV before escaping.
CINCLUSION: It was found that the absorbed dose was approximately 2.5 times more with the p+ core design. The dose to the surrounding PMMA was much less with the p+ core.

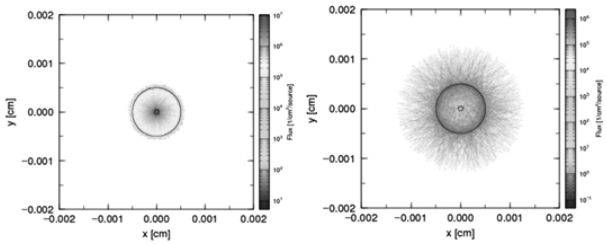

Fig. 1. Ion track of 1.47 MeV alpha particles inside a single SV. Left) p+ core through the centre of the SV. Right) p+ ring around the outside of SV.

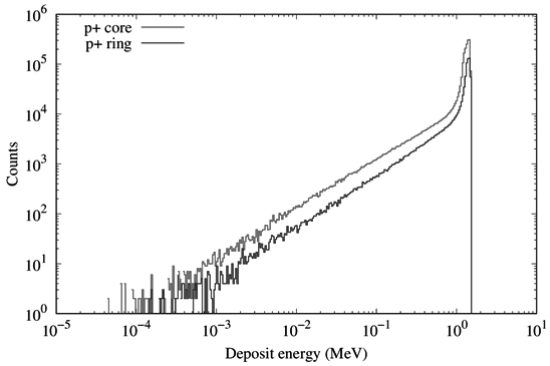

Fig. 2. Deposit energy of 1.47 MeV alpha particles inside the SV for p+ core and p+ ring.

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R2P9-8

PR9-8 Measurement of thermal neutrons and gamma rays in BNCT beam with polymer gel detector

Kenichi Tanaka, Yoshinori Sakurai[1], Tsuyoshi Kajimoto, Aruma Mitsuyasu[1], Shin-ichiro Hayashi[2], Hiroki Tanaka[1], Takushi Takata[1], Gerard Bengua[3], Satoru Endo

Graduate School of Advanced Science and Engineering, Hiroshima University
[1]Institute for Integrated Radiation and Nuclear Science,, Kyoto University
[2]Faculty of Health Sciences, Hiroshima International University
[3]Auckland City Hospital

INTRODUCTION: For the quality assurance of spatial beam component distribution, this study investigates the use of the MAGAT polymer gel detector. The trial to measure thermal neutrons and gamma rays are reported.

EXPERIMENTS: In the experiments for the present study, the MAGAT-type gel detector was infused with LiCl, where the naturally abundant isotope ^{6}Li was used. The ^{6}Li concentrations were set at 0 and 100 ppm. The dimension of the gel detector was 60 × 60 × 60 mm^3. The gel detector was encased in a box made of 5 mm thick acrylic acid resin. The box was set inside a 200 × 200 × 200 mm^3 acrylic acid resin phantom, to simulate a human head in a brain tumor treatment.
The irradiation was performed with the standard epithermal neutron irradiation mode of KUR-HWNIF [1] at 1 MW. The beam size was set to about 120 × 120 mm^2 using the collimator. The irradiation was performed for 110 minutes for 0 ppm and 30 minutes for 100 ppm. The nominal value of the flux at the center of the collimator aperture was 7.07 × 10^6 cm^{-2}s^{-1} and 1.25 × 10^7 cm^{-2}s^{-1} for thermal neutrons and gamma rays, respectively.
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.
The fluence ϕ_j of each component was determined using the following model;

$$S = \begin{pmatrix} S_1 k_1 \\ S_2 k_2 \end{pmatrix} = \begin{pmatrix} a_{11} & a_{12} \\ a_{21} & a_{22} \end{pmatrix} \begin{pmatrix} \Phi_1 \\ \Phi_2 \end{pmatrix} = A \cdot \Phi \qquad (1),$$

$$\Phi = A^{-1} \cdot S \qquad (2),$$

where a_{ij} denotes the sensitivity of the ith gel for the component j, and where k_i denotes the conversion factor from the signal intensity to the absorbed dose. The sensitivity a_{ij} was estimated using the PHITS simulation [2] of the irradiation condition at KUR-HWNIF. Assuming that the signal intensity is proportional to the energy deposition, the ratio of the energy deposition to the beam component fluence at the gel detector regions of 5 × 5 × 5 mm^3 along the central axis was computed and assigned as a representative of a_{ij}. The conversion factor k_i from the signal intensity to the absorbed dose was measured for the irradiation with ^{60}Co gamma rays in Hiroshima University [3].

RESULTS: The response of the gel detector to ^{60}Co gamma rays is shown in Fig. 1. The slopes of the regression lines are 7.8 and 7.1 s^{-1}Gy^{-1}. These correspond to $1/k_i$ in equation (1). Though the signal intensity for KUR irradiation is not shonw here. The estimated fluence is hown in figure 2. The standard deviation is about 2% for gold activation, 10% for TLD. The fluence distribution measured by the gel detector agrees to within 5% to 10% of other estimates. This suggests the potential usability of polymer gel detector in spatial measurement of fluence in BNCT beam.

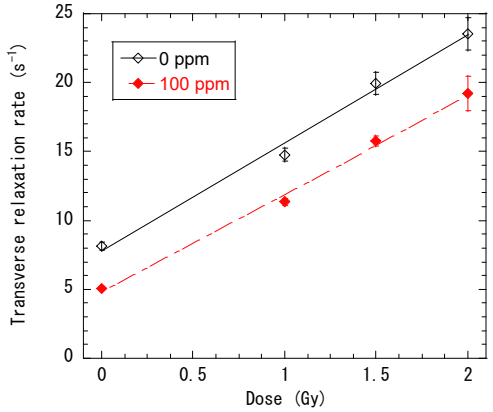

Fig. 1 Response of the gel detector to ^{60}Co gamma rays.

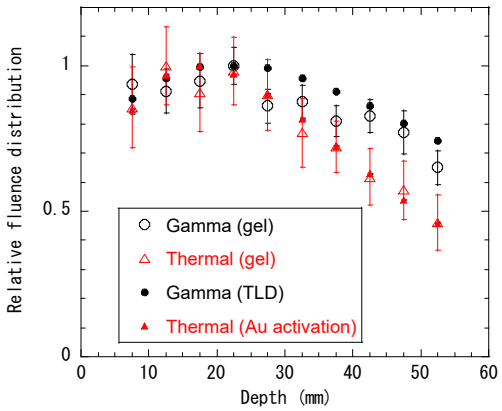

Fig. 2 Relative distribution of fluence estimated.

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PR9-9 Development of two dimensional neutron detector in BNCT irradiation field

H. Tanaka[1], T. Takushi[1], Y. Sakurai[1], S. Uno[2]

[1]Institute for Integrated Radiation and Nuclear Science, Kyoto University
[2] High Energy Accelerator Research Organization

INTRODUCTION:

Boron neutron capture therapy is a radiotherapy that uses alpha rays and lithium nuclei emitted by the reaction of boron-10 and thermal neutrons. Since the accelerator BNCT[1][2] has been covered by insurance, it has been desired to improve the quality assurance method. In particular, since there is no system that can measure the thermal neutron distribution in two dimensions in real time, we are considering the application of a gas electron multiplier (GEM) detector [3] in this study.

While proceeding with the property test at the KUR heavy water neutron irradiation facility, the two-dimensional distribution was measured by the gold activation method in order to evaluate the validity of the thermal neutron distribution obtained by a GEM detector.

EXPERIMENTS:

The gold film was irradiated under the condition of a collimator diameter of 12 cm in the mixed mode of the KUR heavy water irradiation facility. The size of the gold film was 10 cm square, and it was installed covering a quarter of the collimator.

After irradiation for 15 minutes, the activated gold film was placed on an imaging plate (IP) and exposed. A small gold foil with a thickness of 50 μm was installed at positions 42 mm from the center of irradiation, and used as a reference foil for standardizing the thermal neutron flux.

After irradiation, the saturated radioactivity of gold foil was measured by a germanium detector, and thermal neutrons were evaluated. The photo-stimulated luminescence (PSL) value of an IP at the same point was normalized by the thermal neutron flux.

RESULTS:

Fig. 1 shows the result of the two-dimensional distribution of PSL values. The fig. 2 shows the result of the thermal neutron flux normalizing the PSL value in the r-axis direction.

It was confirmed that the collimator was able to shield about two digits. In the future, a GEM detector will be installed directly under the collimator to compare the thermal neutron distribution.

CONCLUSION:

Two-dimensional thermal neutron distribution was measured by the combination between a gold film and an IP. This method is not able to detect thermal neutron in real time. And the procedure is time consuming. However, position resolution is good because it is depended on an IP position resolution around 50 μm. In this time, two-dimensional thermal neutron distribution was evaluated to compare with the experiment results of a GEM detector.

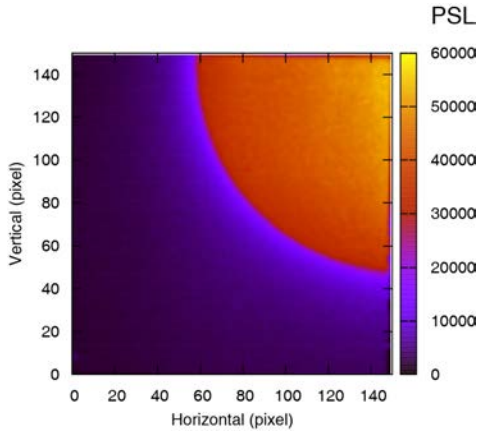

Fig. 1. The photo-stimulated luminescence (PSL) distribution exposed with activated gold film.

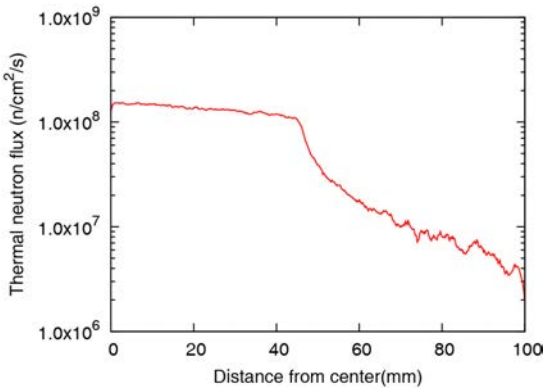

Fig. 2. Normalized thermal neutron distribution.

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PR9-10 Study of Neutron Fluence and Gamma ray Distribution Measuring using Thermoluminescence Slabs

K. Shinsho, R. Oh, M. Tanaka, S. Yanagisawa, H. Tanaka[1], T.Takata[1], G. Wakabayashi[2], Y. Koba[3]

Graduate School of Human Health Science, Tokyo Metropolitan University

[1] Institute for Integrated Radiation and Nuclear Science, Kyoto University

[2] Graduate school of Science and Engineering Research, Kindai University

[3]Center for Radiation Protection Knowledge, QST-NIRS

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 can irradiate a cancer cell at the molecular level selectively. However, dosimetry techniques in mixed neutron-gamma fields have not been established yet. Therefore, we focused on reusable two-dimensional thermoluminescence dosimeter (2D-TLD). The 2D-TLD which we used is thermoluminescence (TL) phosphor Cr doped Al_2O_3 ceramic plate (2-D Al_2O_3: Cr TLD).[1] 2-D Al_2O_3: Cr TLD that can acquire radiation images with high spatial resolution. In addition, it has high solidity and can be used under water. We are investigating neutron imaging using 2-D Al_2O_3: Cr TLD and Cd neutron-gamma ray converter technique in mixed neutron-gamma fields. Furthermore, we are conducting research on γ-ray imaging in mixed fields using TL characteristics of BeO ceramic plates. And then, we have found that the Cd neutron-gamma ray converter lets the sensitivity of the neutron increase selectively, and we can acquire theneutron images easily without dis-crimination of gamma ray. In this study, the thermal neu-tron fluence distribution was measured using a large-area 2-D Al_2O_3: Cr TLD with Cd converter, and the γ-ray measurement using BeO ceramic plates was performed in 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% was used for measurements of thermal neutron fluence distribution. The bulk density of the plates was 3.7g·cm^{-1}. The dimensions used for the glow curve measurements were 10 × 10 × 0.7 mm^3 and 80 × 80 × 0.7 mm^3. The concentration of Cr_2O_3 in the present study was 0.05 wt%. BeO ceramic plates（Thermalox995) of Materion Corporation, which was composed of BeO > 99.5 wt %, Si 0.186 wt%, Mg 0.092 wt%, was used for γ-ray measurement in mixed neutron-gamma field. The bulk density of the plates was 3.02 g·cm^{-1}. The dimensions used for the glow curve measurements were 10 × 10 × 1.0 mm^3. The assumed irradiation fields are the mixedneutron 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·s^{-1}. The 2-D TL measurement system consists of a CMOS camera, 80×80 mm^2 heater, and a dark box. After expo-sure, the TL slabs were heated to 400 ℃ for 5 min. The TL images were captured using a CMOS camera equipped with a thermal cut filter.

RESULTS: Figure 1 shows the relationship between the neutron reaction rate of Cd in KUR and the TL amount of Al_2O_3: Cr. From Fig.1, it became clear that the neutron reaction rate and the amount of TL are represented by quadratic functions. Since the nuclear reaction cross section of Cd to neutrons is constant, it was suggested that by measuring the TL amount of Al_2O_3: Cr, it may be possible to investigate the presence or absence of changes in the neutron energy spectrum during periodic inspections. Figure 2 shows arrangement of TL slabs irradiation arrangement for both TL imaging and the TL image of the slab. It was shown that high spatial resolution thermal neutron distribution measurement is possible.In γ-ray measurement in mixed neutron-gamma field using BeO ceramic plates, the results of the BeO ceramic plate showed good agreement with the TLD (UD-170LS) of Panasonic Corporation.

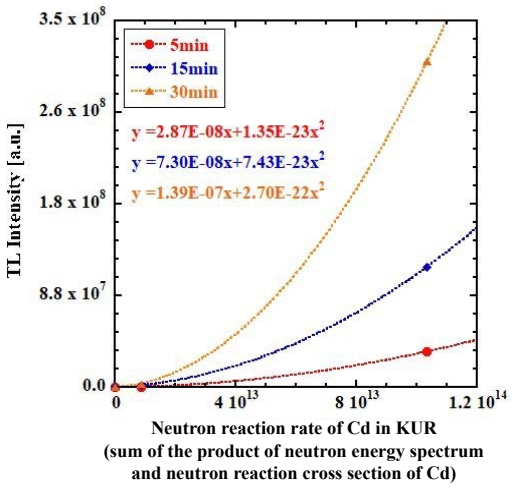

Fig.1 relationship between the neutron reaction rate of Cd in KUR and the TL amount of Al_2O_3: Cr.

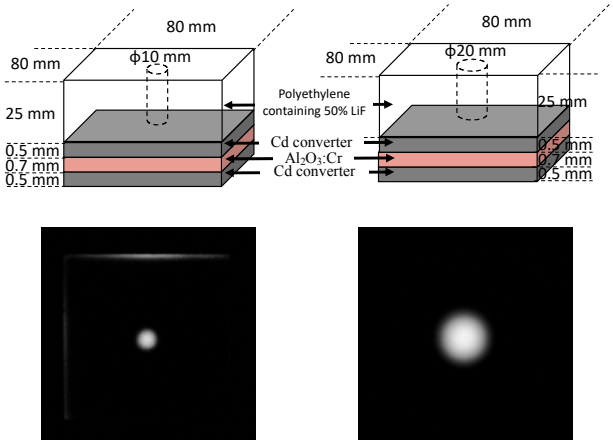

Fig.2 Upper: Arrangement of TL slabs (Al_2O_3: Cr using Cd converter) irradiation arrangement for TL imaging. Lower: TL image of the slab.

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R2P9-11

PR9-11 Development and evaluation of 3D gel dosimeter for the measurement of dose distribution in BNCT

S. Hayashi, Y. Kakimoto[1], T. Takata[2], M. Suzuki[2] and Y. Sakurai[2]

Department of Clinical Radiology, Hiroshima International University
[1]Graduate School of Engineering, Kyoto University
[2]Institute for Integrated Radiation and Nuclear Science, Kyoto University

INTRODUCTION: Irradiation field for boron neutron capture therapy (BNCT) is the mixed field of thermal neutrons, epi-thermal neutrons, fast neutrons, and gamma rays, so the discrimination estimation is necessary for each dose component in the irradiation field characterization. The point-wise estimation using activation foil, thermo-luminescent dosimeter (TLD), ionization chamber, etc., is mainly performed at present, but two-dimensional (2D) estimation is also needed. As part of the advancement in characterization for BNCT irradiation field, a study was performed for the 2D component-discrimination estimation using a radiochromic gel dosimeter base on polyvinyl alcohol - glutaraldehyde - iodide (PVA-GTA-I) [1].

MATERIALS AND METHODS: Three-layered planer gel dosimeter was prepared, as shown in Figure 1 (left). Each planar gel dosimeter was 10 cm in width, 10 cm in length and 0.3 cm in thickness. The first layer (Gel-1) was a PVA-GTA-I gel dosimeter containing boric acid ($B(OH)_3$) of natural abundance for boron-10 or lithium sulphate (Li_2SO_4) of enrichment for lithium-6 as the sensitizers for thermal neutrons. The second layer (Gel-2) was a normal PVA-GTA-I gel dosimeter without the sensitizers. The third layer (Gel-3) was a PVA-GTA-I gel dosimeter consisting of heavy water instead of light water. For fast neutrons, the recoiled protons or deuterons due to the elastic scatter of hydrogen or deuterium mainly contribute to dose. For thermal neutrons, the nuclear reactions of boron-10 or lithium-6 mainly contribute to dose.
The response characteristic for gamma rays was estimated using Co-60 gamma-Ray Irradiation Facility. The response characteristic for neutrons was estimated by the irradiation experiments using three irradiation modes for thermal, epi-thermal, and mixed neutrons at Heavy Water Neutron Irradiation Facility (HWNIF) of KUR [2], and by the simulation calculation using Particle and Heavy Ion Transport-code-System (PHITS) [3].

The verification experiment for 2D component-discrimination estimation was performed using the mixed neutron irradiation mode of KUR-HWNIF. The size of the collimator aperture was 12 cm in diameter, and the three-layered planer gel dosimeter was placed at the collimator exit to hang on the aperture edge (*e.g.*, Figure 1 (right)).

The irradiated gel dosimeters were scanned using a scanner and the transmission images were obtained. The transmission images were analyzed using a Python program, the net optical density (net OD) was calculated from the analogue digital conversion (ADC) values, and the response characteristic was estimated.

RESULTS AND DISCUSSION: From the response characteristic experiment for gamma rays, the high linearity was confirmed between the response and dose from 0 to 20 Gy. From the response characteristic experiment for neutrons, the response curve to dose-weighted linear energy transfer (LET) was obtained. Using this response curve, the correction factors regarding LET-dependency for the recoiled protons, recoiled deuterons and the thermal neutron reactions with boron-10 and lithium-6, were obtained.

From the verification experiment for 2D component-discrimination estimation, the 2D dose distributions were obtained. It was confirmed that the 2D distribution for thermal neutrons were relatively in close agreement with the estimated result by activation method using gold thin film. The absolute value by the gel dosimeter was almost two times larger compared with the activation method. It is thought that the components scattered in the gel dosimeter contributed additionally. For gamma rays, the 2D dose distribution was in good agreement with the estimated result by TLD.

CONCLUSION: The feasibility for the 2D component-discrimination estimation using the three-layered planer gel dosimeter was confirmed. In future, the sensitivity improvement and thickness reduction of this gel dosimeter will be studied. Also, the dependency for LET will be precisely estimated.

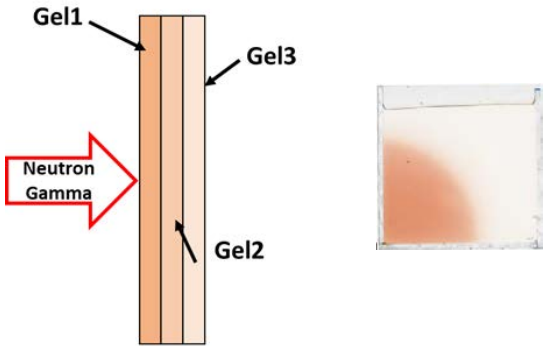

Figure 1 (left) Schematic of three-layered planer gel dosimeter. (right) An irradiated gel sample.

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R2P9-13

PR9-12 Establishment of beam-quality estimation method in BNCT irradiation field using dual phantom technique (IV)

Y. Sakurai, N. Kondo, T. Takata, H. Tanaka and M. Suzuki

Institute for Integrated Radiation and Nuclear Science, Kyoto University

INTRODUCTION: Development in several types of accelerator-based irradiation systems for boron neutron capture therapy (BNCT) is underway. 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 are 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 [1].

In 2020, verification experiments for the dual phantom technique were performed using Heavy Water Neutron Irradiation Facility installed in Kyoto University Reactor (KUR-HWNIF) as in the previous year. Biological experiments were performed using the solid phantoms, which were made based on the simulation results.

MATERIALS AND METHODS: One of the dual solid phantoms was made of polyethylene with natural lithium fluoride for 30 weight percent (LiF-polyethylene phantom), and the other phantom was made of polyethylene with 95%-enriched lithium-6 fluoride for 30 weight percent (^{6}LiF-polyethylene phantom).

Vials containing one kind of human brain cell, such as U87ΔEGFR, were placed at the surface, 2-cm depth, 5-cm depth and 8-cm depth in the phantoms on the center axis of the beam line. Cell growth assay was performed for the irradiated cells.

The neutron flux and gamma-ray dose rate were measured using activation foils and thermo-luminescent dosimeter, respectively. The depth dose distributions for the thermal neutron, fast neutron and gamma-ray components were determined based on the simulation calculation results normalized referring to the measured values.

Figure 1 shows a photograph of the experimental setup. The epi-thermal neutron irradiation mode was used.

RESULTS: Figure 2 shows the depth distributions of the cell survival for the BPA administration group, BPA(+), and the non-administration group, BPA(-), in the LiF-polyethylene phantom. For BPA(+) group, the concentration of boron-10 was 25 ppm. The difference for the cell survival was shown between BPA(+) group and BPA(-) group. Due to the neutrons thermalized in this phantom, the cell survival for BPA(+) group was lower than that for BPA(-) group.

Figure 3 shows the depth distributions of the cell survival in the ^{6}LiF-polyethylene phantom. In this phantom, most of the thermalized neutrons were absorbed by ^{6}LiF. So, the cell-survival distributions for BPA(+) group and BPA(-) group were almost the same.

CONCLUSION: The assay results will be more analyzed in association with the data of the depth dose distribution for the thermal neutrons, fast neutrons and gamma-rays.

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

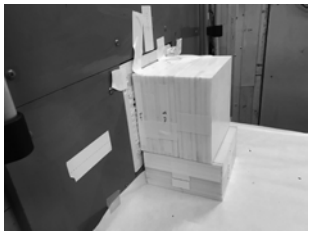

Fig. 1. A photograph of the experimental setup.

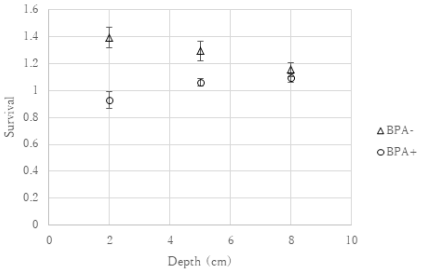

Fig. 2. Depth distributions for the cell survival in the LiF-poly phantom.

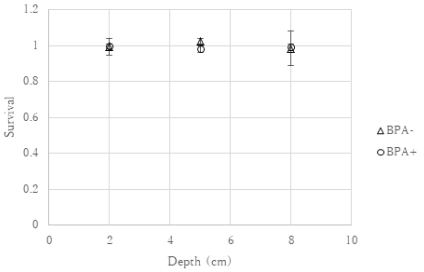

Fig. 3. Depth distributions for the cell survival in the ^{6}LiF-poly phantom.

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R2P9-14

PR9-13

Development of a prompt gamma-ray imaging detector using an 8 x 8 arrayed $LaBr_3$(Ce) scintillator for BNCT

K. Okazaki[1], T. Takushi[1], Y. Sakurai[1], H. Tanaka[1]

[1]*Institute for Integrated Radiation and Nuclear Science, Kyoto University*

INTRODUCTION: New boron compounds have been investigated with small animal studies for improving boron neutron capture therapy (BNCT). To evaluate the boron dose, the ^{10}B concentration in animal body is needed. Usually, the information on ^{10}B concentration in the animal body has been evaluated using a high-purity germanium detector with prompt gamma-ray analysis[1] or the induced coupled plasma. However, these methods can not measure ^{10}B concentration in real time. A real-time detector has been developed to obtain the information on ^{10}B concentration in basic research of BNCT. An 8 x 8 arrayed $LaBr_3$(Ce) scintillator and Multi Pixel Photon Counter (MPPC) were used to acquire information on ^{10}B concentration by measuring prompt gamma rays emitted from the reaction between thermal neutrons and ^{10}B. This detector system needs the energy resolution of 6.5 % for 511 keV gamma-ray to discriminate between prompt gamma rays of 478 keV and annihilation gamma rays of 511 keV as background events. This report presents the characteristics about energy resolution of this detector system.

EXPERIMENTS: This detector system consists of an 8 x 8 arrayed $LaBr_3$(Ce) scintillator, an 8 x 8 arrayed MPPC, 64 channels amplifiers-shapers, and 64 channels analog digital convertors. The size of each scintillator was 5 mm x 5 mm x 10 mm and the light yield was 63 photons/keV. There were reflectors made from $BaSO_4$ between scintillator pixels. The effective active area of one channel of the MPPC was 6 mm x 6 mm, and the MPPC had 14400 pixels in total. The scintillator was placed in front of the MPPC. Energy calibration were performed using common isotopes (^{22}Na, ^{57}Co, ^{133}Ba, and ^{137}Cs). In order to confirm the characteristics of this system, particle and heavy ion transport code system (PHITS) was used.

RESULTS: Fig. 1 shows the relationship between ADC channels and gamma-ray energy. Theoretically, saturation ADC channel would be corresponding to around 230 keV. Furthermore, we calculated events that Compton scattering occurred in one pixel of the scintillator and entered other pixels. The events were 26.5 % of total events with PHITS. In the experiment data, the events those were deposited energy in two or three pixels were corresponding to the Compton scattering. The events were 26.9 % of total events in the experiment. The gamma-ray spectra with an ^{22}Na were broadened because summed signals from two or three pixels of the MPPC were large because of the distribution of deposit energy without the saturation. This came from Compton scattering. After the spectrum was calibrated, the energy resolution for 511 keV peak was 5.0 % as shown in Fig. 2.

CONCLUSION: The developed prompt gamma-ray imaging detector measured gamma-ray spectra; however, gamma-ray spectra obtained from raw data had broadened peaks. We corrected events caused by Compton scattering because the MPPC acquired higher ADC channels. As a result, after this detector system was calibrated, it had about 5.0 % energy resolution for 511 keV peak. This is sufficient to discriminate between 478 and 511 keV gamma rays.

Acknowledgements
This study was partially supported by JSPS KAKENHI Grant Number JP25282155 and 16H03193.

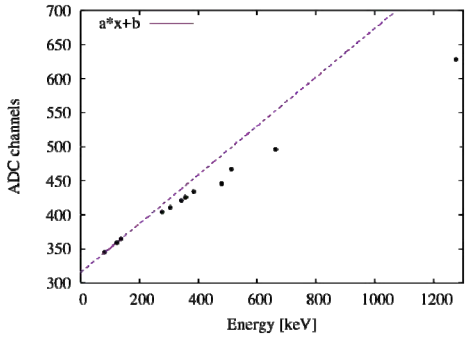

Fig. 1. Relationship between ADC channels and gamma ray energy.

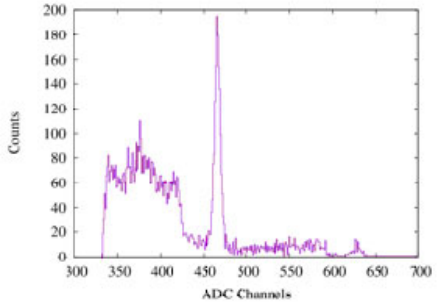

Fig. 2. Gamma-ray spectrum at the center of the MPPC after the correction of Compton scattering events.

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R2P9-15

PR9-14 Development of Fiber-reading Radiation Monitoring System with an Optical Fiber and Red-emitting Scintillator at the ^{60}Co Radiation Facility

S. Kurosawa[1,2], H. Tanaka[3], S. Kodama[2,] Y. Kurashima, Akihiro Yamaji

[1]*New Industry Creation Hatchery Center, Tohoku University*
[2] *Institute for Materials Research Tohoku University*
[3] *Institute for Integrated Radiation and Nuclear Science, Kyoto University*

INTRODUCTION: Decommissioning reactors at the nuclear power plant with safety is a big issue in Japan, and a real-time dose-rate monitor in the extremely high radiation dose condition is required for the above application. We have developed this monitor consisting of a scintillator, optical fiber and CCD spectrometer, and last year we succeeded in demonstrating the monitor in the ^{60}Co radiation facility of the Institute for Integrated Radiation and Nuclear Science, Kyoto University, using Cs_2HfI_6 (CHI) scintillator with an emission band of around 700 nm [1].

CHI has other attractive points such as high light output (over 60,000 photons/MeV), high effective-atomic number (over 60) and no afterglow (less than 1% within 1s), while this material cannot be applied to alpha-ray detection due to the package for the hygroscopic nature[1,2]. Thus, we demonstrated the monitor using novel materials with red/infrared emission and without hygroscopic nature in the ^{60}Co radiation facility.

Since the size of the scintillation sample should be small as well as a few mm^3 due to the fiber diameter and space limitation in the power plant, so that a high effective atomic number is required. One of the candidate materials is Yb-doped $La_2Hf_2O_7$ (Yb:LHO), because this atomic number is 64. Moreover, Yb-doped samples are expected to show an infrared sharp peak around 970 nm originating from Yb^{3+} of 4f-4f transition. Although Yb:LHO has a high melting point of over 2400C, we grew this crystal by the Core-Heating (CH) method we have developed as a novel crystal-growth technique in 2020[3].

EXPERIMENTS: We fabricated a Yb:LHO sample grown by the CH method, and scintillation properties such as transmittance spectrum and emission spectrum excited by UV and X-ray photons.

The demonstration was performed at the ^{60}Co Gamma-ray Irradiation Facility with an activity of ~100 TBq, and the gamma-ray air dose rates at certain points on the experimental apparatus were estimated based on the absorbed gamma-ray doses for water measured by Sato et al.[4]. The monitor and set up at this time were the same as that in last year; an optical fiber (S.600/600B, Fujikura) had a length of 20 m and a pure SiO2core diameter of 600 ± 30μm using an optical grease. The scintillation light was transmitted through the fiber and measured with a CCD spectrometer (Blue-UVNb, StellarNet).

RESULTS: We succeeded in growing the transparent Yb:LHO sample for the first time by the CH method as shown in Fig. 1, even the high-melting point material, and its transmittance was ~50% at 870 nm. The sharp emission peak was observed at 974 nm excited by X-rays and UV photons, while CHI had broad emission peak at around 700nm. The noises were generated in the onptical fiber had mainly photons below 600 nm, and some noises have remained even at 700 nm, still.

Figure 2 shows the sicintillation intensities with the CCD at several dose-rate positions for Yb:LHO and Cr:α-Al_2O_3 as the reference sample. The intensity of Yb:LHO had lower intensity (1% of CHI) than those of Cr:α-Al_2O_3 (10% of CHI), and dose-rate dynamic range was narrow than the others. Nevertheless, Yb:LHO had a good Signal-to-Noise ratio (S/N) by 5 – 100 times than the others had, depending on noise condition (dose-rate condition for the optical fiber). This good S/N is due to small noise at ~970 nm and sharp peak, and the wavvelenth sensitive detector is effective to remove the noise. We succeeded in development of the novel material, Yb:LSO, with good S/N for the dose monitor.

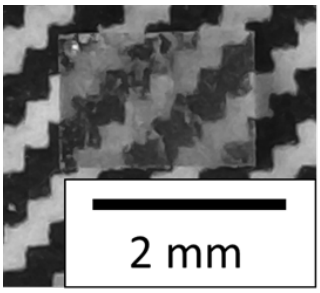

Fig. 1. Pphotograph of the Yb:LHO sample after cut and polishing.

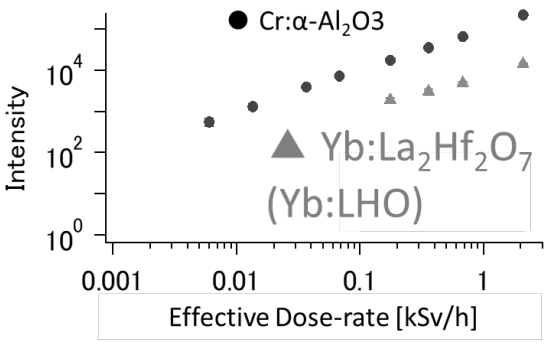

Fig. 2. The scintillation signal intensity of CHI cement and ruby cement specimens, respectively, as a function of effective dose rate.

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[3]N. Sato et al., (2008), [www.rri.kyoto-u.ac.jp/gamma/doseratesheet.pdf]
[4]Y. Kurashima *et al.*, Cryst. Growth Des., **21** (2021) 572–578.

R2P9-16

PR9-15 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

S. Komura, T. Mizumoto, Y. Sakurai[1], T. Takata[1], T. Tanimori[2] and A. Takada[2]

Fukushima SiC Applied Engineering Inc.
[1]Institute for Integrated Radiation and Nuclear Science, Kyoto University
[2]Graduate School of Science, Kyoto University

INTRODUCTION: Boron neutron capture therapy (BNCT) is one of the promising cancer treatment methods. However, a good method for real-time monitoring of therapeutic effects during BNCT is not yet available. The main reason is that it is difficult to accurately know the concentration of boron-containing drugs in tumor tissue and healthy tissue. If we can image the 478 keV prompt gamma rays emitted from the $^{10}B(n,\alpha)^{7}Li$ reaction occurring in the patient's body, we can confirm the concentration of these drugs in each site. Although detectors such as SPECT cameras and Compton cameras have been proposed to obtain images of the 478 keV prompt gamma rays, each of them has its own weaknesses and is not yet ready for practical use [1].

To overcome this situation, we have been developing an advanced Compton camera, the Electron Tracking Compton Camera (ETCC), which can uniquely determine the direction of arrival of sub-MeV/MeV gamma rays for each event by measuring information on the recoil direction of electrons. An ETCC is a complex detector with two sub-detectors: a time projection chamber TPC (Compton scatterer and recoil electron detector) and a scintillation camera (scattered gamma-ray absorber). In our previous studies, we demonstrated that our ETCC provides high contrast images of 478 keV prompt gamma rays emitted from a large water phantom with a ^{10}B concentration of 2000 ppm using E-3 Neutron Guide Tube (E-3) at Kyoto University Reactor (KUR) [2]. In the present study, two experiments were performed to confirm the performance of our ETCC under more realistic conditions.

EXPERIMENTS AND RESULTS:

Exp. 1: We performed prompt gamma-ray imaging studies using a boron-containing solution in a 4.5 mm diameter tube that simulates a small tumor tissue. The experimental setup is shown in Fig. 1. The tube was placed just above the ETCC and was irradiated with thermal neutrons at E-3 at the 1-MW operation. The ETCC measured the prompt gamma rays emitted from the solution during the irradiation time and reconstructed a back-projection image. As a result, high quality images with a spatial resolution of less than 1 cm were obtained. This experiment was conducted on four tubes with different ^{10}B concentrations (0, 500, 1000, 2000 ppm) to investigate the relationship between ^{10}B concentration and image brightness. These results are currently being analyzed.

Exp. 2: We also carried out performance tests in a BNCT environment, a strong gamma-ray environment where conventional Compton cameras are disturbed by random coincidence noise, using Heavy Water Neutron Irradiation Facility of KUR at the 1-MW operation. As shown in Fig. 2., a 10-cm cubic solution with a ^{10}B concentration of 230 ppm was placed in the irradiation room. The prompt gamma rays emitted from the solution and passing through the cylindrical hole in the ceiling entered the ETCC. The obtained image precisely shows the position of the solution even in the huge background radiation environment. This experiment was performed twice by changing the position of the solution.

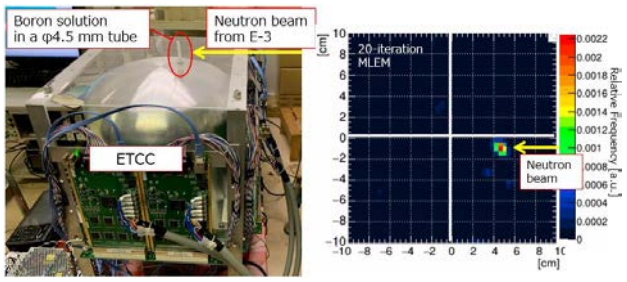

Fig. 1. Photograph and obtained image of 478 keV prompt gamma rays from ^{10}B 500 ppm tube

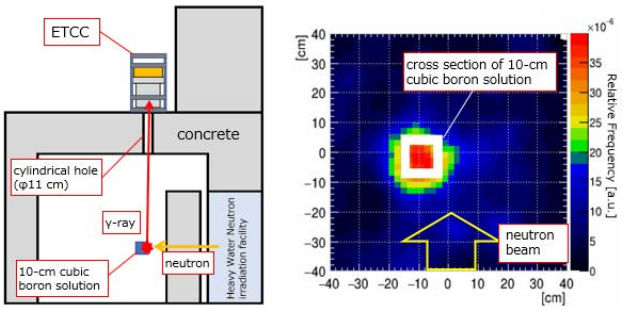

Fig. 2. Schematic view and obtained image of 478 keV prompt gamma rays from 10-cm cubic boron solution

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[2] T. Mizumoto, KURNS Progress Report 2019 (2020).

PR9-16 Feasibility study for quality assurance and control for irradiation field in BNCT

S. Nakamura[1,2], M. Takemori[1,3], K. Iijima[1], S. Imamichi[4], H. Nakayama[1,3], T. Chiba[1,3], M. Masutani[5], Y. Sakurai[6], H. Tanaka[6], T. Takata[6], M. Suzuki[6], H. Igaki[7] and H. Okamoto[1,2]

[1]Department of Medical Physics, National Cancer Center Hospital.
[2]Devisoin of Research & Development for Boron Neutron Capture Therapy, National Cancer Center Exploratory Oncology Research & Clinical Trial Center.
[3]Department of Radiological Science, Graduate School of Human Health Sciences, Tokyo Metropolitan University.
[4]Devision of Cellular Signaling, National Cancer Center Research Institute.
[5]Department of Molecular and Genomic Biomedicine, Graduate School of Biomedical Sciences, Nagasaki University.
[6]Institute for Integrated Radiation and Nuclear Science, Kyoto University
[7]Department of Radiation Oncology, National Cancer Center Hospital

INTRODUCTION: In boron neutron capture therapy (BNCT), the neutron irradiation to the patient is performed after a sufficient accumulation of ^{10}B to tumor cells because the treatment efficacy is based on the ^{10}B(n,α)^{7}Li reaction. It is important for the radiation therapy that the delivered dose is verified as a quality assurance before the treatment. However, it is a difficult for the existing detector that the three dimensional dose distribution is measured in BNCT with considering a non-uniformity of ^{10}B on the actual patient. Thus, this study focuses on the gel-dosimeter for the three dimensional dosimetry in BNCT. There are some reasons. First, the gel-dosimeter can mix a few ^{10}B with keeping the dose response. Second, the gel-dosimeter is generally used for the three dimensional dosimeter. Third, the gel-dosimeter is not expensive, and the quality assurance using the gel-dosimeter may not affect the medical costs in BNCT. Therefore, this study aims to evaluate the dose response dependence of the gel-dosimeter which contains a certain boron-10 density in BNCT.

EXPERIMENTS: The experiment was performed in the Heavy Water Neutron Irradiation Facility (Kyoto University Reactor, KUR). The KUR was operated at a power of 1 MW. The neutrons were delivered to the gel-dosimeter which composed of water, PVA, KI, KCl, Gellan gum, Fructose, and boron solutions. The boron-10 density of the gel-dosimeter was 0, 25, 50, and 75 ppm. The irradiation time was set to 0, 5, 10, 20, 30 min in each gel-dosimeter. After the neutron irradiation, the dose response in each gel-dosimeter was read by an optical scanner. The dose response dependence in each gel-dosimeter was evaluated. In our previous experiment, the mixing time for the gel-dosimeter was 15 min, and the dose response in each gel-dosimeter then had variations. Thus, the mixing time for this experiment was set to 60 min.

RESULTS: Figure 1 (a) shows the dose response of the gel-dosimeter for each boron-10 density in this experiment. Additionally, Fig. 1 (b) shows it derived from our previous study. As shown in Fig. 1, in this experiment, the dose responses against the irradiation time was better than those in our previous study.

Discussion and Conclusion: This study investigated that the gel-dosimeter has a sufficient sensitivity for the reaction of ^{10}B(n,α)^{7}Li in BNCT although the boron-10 density expects very low (25 ppm, etc…). According to fig. 1, the dose response in the gel-dosimeter increased with the boron-10 density and the irradiation time, and it seems that the dose response can evaluate the actual dose delivered to the gel-dosimeter. Although linearity of the dose response was not better in our previous study, it is improved in this study. It relates to a creating method of the gel-dosimeter. In this experiment, the mixing time is longer than that in our previous study. Therefore, we are able to fix the creating method for the gel-dosimeter. We will try to measure three dimensional dose with using the gel-dosimeter in the future work.

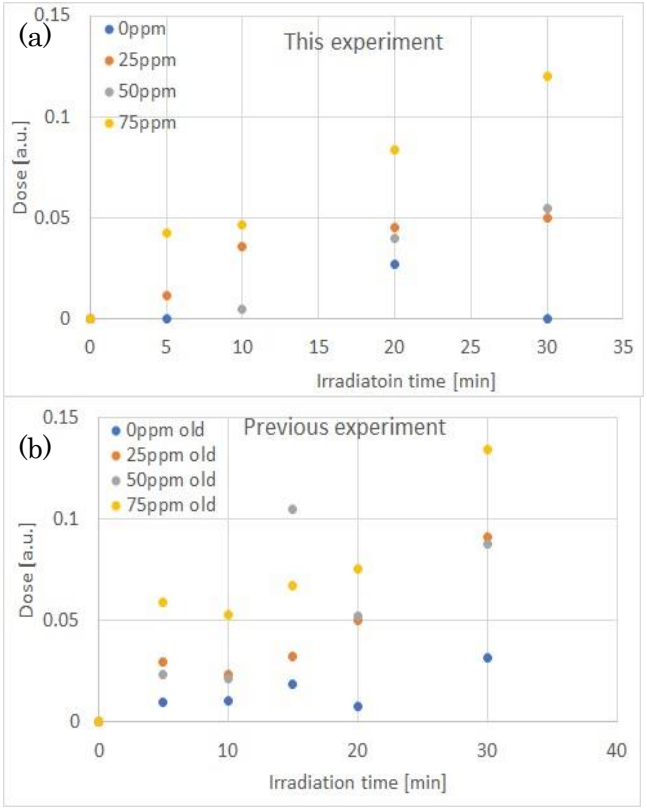

Fig. 1. Dose response dependence in the gel-dosimeter. (a): This experiment, (b): our previous study

PR9-17 Evaluation of thermal neutron irradiation field for semiconductor device irradiation

H. Tanaka[1], T. Takushi[1], N. Matsubayashi[1] , Y. Sakurai[1]

[1]*Institute for Integrated Radiation and Nuclear Science, Kyoto University*

INTRODUCTION: At the BNCT irradiation field, electronic devices with semiconductor such as neutron monitors, patient monitor cameras, and implantable pacemakers may be irradiated. If a semiconductor device contains a nuclide having a large capture cross-section with respect to thermal neutrons, charged particles are generated in the semiconductor device when irradiated with thermal neutrons. The charged particle generates an electron-hole pair in its trajectory, and this charge becomes a noise current in the semiconductor device. There is a concern that the semiconductor device may malfunction due to this electric current. In addition, semiconductor devices are becoming finer year by year, and the signal current is becoming smaller. That is, it tends to be more sensitive to noise. To date, there are few data on semiconductor damage to thermal neutrons. In this study, we evaluated the thermal neutron irradiation field necessary for investigating the effects of thermal neutrons on a semiconductor device.

EXPERIMENTS: The heavy water neutron irradiation facility (HWNIF) slows down the fast neutrons generated in the reactor core to thermal neutron energy with heavy water and aluminum. In order to evaluate the effect of thermal neutrons on semiconductor devices, an irradiation field that changes only the thermal neutron flux without changing the epithermal and fast neutrons is required. The HWNIF is equipped with a aperture changeable cadmium filter behind the heavy water moderator. It is possible to change the thermal neutron flux. Further, the semiconductor device is installed on the substrate, and in order to observe the semiconductor error in real time, it is necessary to install a measuring instrument on a large career. The thermal neutron flux on the bismuth surface using an irradiation rail have been reported [1], however the combination of a large career and a collimator has not been evaluated. Here, the thermal neutron flux at the outlet of the collimator in diameter of 300 mm x 300 mm was evaluated using the gold activation method.

RESULTS: Fig. 1 showed the result of the thermal neutron flux at the center of the collimator outlet when the aperture size of cadmium filter was changed. Fig.2 showed the cadmium ratio. The thermal neutron flux at the position of the bismuth surface without a collimator was reported as 1 x 10^9 (n/ cm^2/s) for a 600 mm cadmium filter opening. However, thermal neutron flux with collimator was about 60%. Since it was far from the bismuth surface and was collimated. In addition, the cadmium ratio tended to increase because hydrogen was contained in the material of the collimator.

CONCLUSION: In the HWNIF, the relationship between the aperture size of cadmium filter and the thermal neutron flux was derived in the irradiation field combining a large career and a collimator. In the future, we plan to irradiate semiconductor devices based on this information.

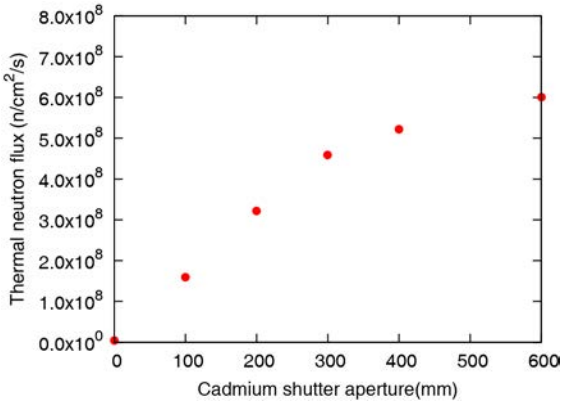

Fig. 1. Relationship between thermal neutron flux at reactor power of 1MW and the aperture size of cadmium filter.

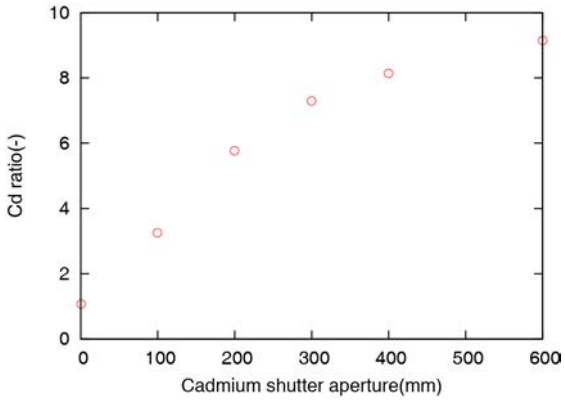

Fig. 2. Relationship between the cadmium ratio and the aperture size of cadmium filter.

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PR9-18 Optimization of Bolus Shape for Boron Neutron Capture Therapy — Examination Using Simple Shaped Phantom for Experimental Verification —

T. Takata, H. Tanaka, A. Sasaki[1], Y. Sakurai and M. Suzuki

Institute for Integrated Radiation and Nuclear Science, Kyoto University
[1]Graduate School of Engineering, Kyoto University

INTRODUCTION: In Boron Neutron Capture Therapy (BNCT), epithermal neutron beam has been utilized to treat a deep-seated tumor due to its high penetration ability based on thermalization within a patient body. However, thermal neutron buildup causes dose deficiency in a case where a tumor extends to a vicinity of patient surface. In such a case, a bolus consisting of a hydrogen-rich material has been utilized to improve the dose distribution. In present clinical BNCT, a bolus with a uniform thickness and a simple shape has been adopted. Aiming to increase a tumor dose more aggressively, the authors have studied an optimizing method of the bolus shape. A calculation of the bolus optimization using a simple shaped phantom was conducted prior to an experimental verification, and the result is described in this report.

MATERIALS AND METHODS: The optimization was performed according to the method previously reported by the authors [1]. A cylinder phantom assuming a head-and-neck case with a parotid gland cancer was used in this calculation. The detailed calculation procedure was as follows.

Phantom geometry modelling: A cylindrical water phantom with the height of 20 cm and diameter of 20 cm was modeled for the calculation (Fig. 1). A spherical volume with 4-cm diameter of which the center was located at 2-cm depth from the side surface of the cylinder, was defined as a planning target volume (PTV), assuming a parotid gland cancer extended to a subcutaneous region. In addition, a tubular volume with 3-cm diameter along the center axis of the cylinder was defined as an organ at risk (OAR), assuming a mucosal tissue of oral cavity and pharynx. These structures were prepared as a voxel model for the following transport simulation.

Neutron and photon transport calculation: The transport calculations were performed by using a Monte Carlo transport calculation module, SeraMC, implemented in SERA; the treatment planning system for BNCT [2]. An epithermal neutron beam of KUR heavy water neutron irradiation facility with an aperture of 12-cm diameter was assumed in the calculations [3]. The phantom was placed in contact with the aperture so that the beam entered the phantom from its side and passed through the center of the PTV. The results of the transport calculation were analyzed by using a dose evaluation module of SERA.

Bolus formation: A bolus forming area was limited within a surface area in a 5-cm distance from the beam incident point. Control points of bolus thickness were assigned to the representative voxels in the area by using a 1.5 × 1.5 × 1.5 cm^3 mesh-grid down sampling method for reduction of the calculation time. A bolus region was generated in the geometry by using an inverse-distance weighted interpolation of the thicknesses assigned to each of control points.

Optimization calculation: An objective function was defined as the ratio of the minimum PTV dose to the maximum OAR dose. The bolus thicknesses at the control points were adjusted to maximize the dose ratio by using a steepest gradient descent method.

RESULTS: The control points were assigned to 37 surface voxels by using the method described above. The iterative calculation was stared from bolus thicknesses of zero at all control points. The optimization result is shown in Fig.1 and Fig. 2. Increase and convergence in the dose ratio with increase in number of iterations were confirmed as Fig. 2. Figure 1 also shows the resulting bolus shape at the 3rd iteration, at which the objective function was almost converged. Growth of the bolus was observed around the area where the PTV reached just under the surface. We plan to create the bolus with the optimized shape by using 3D printer. Then, we will verify improvement of the dose distribution experimentally based on the calculation result.

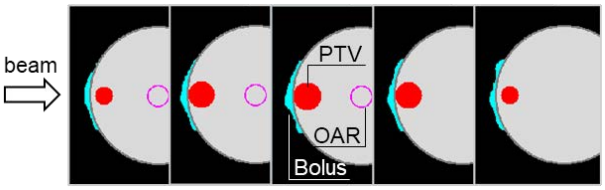

Fig. 1. Phantom geometry with the optimized bolus.

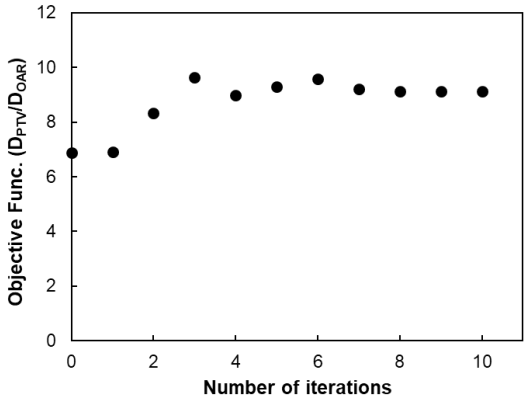

Fig. 2. Change in the objective function according to an iterative calculation.

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R2P9-20

PR9-19 Verification of the annealing capability of boric acid-infused PVA-GTA-I gel dosimeter

H. Yasuda[1], JE. Taño[1,2,3], CAB. Gonzales[1], Y. Sakurai[4]

[1]*Department of Radiation Biophysics, RIRBM, Hiroshima University*
[2] *Graduate School of Biomedical and Health Sciences, Hiroshima University*
[3] *Phoenix Leader Education Program for Renaissance from Radiation Disaster, Hiroshima University*
[4] *Institute for Integrated Radiation and Nuclear Science, Kyoto University (KURNS)*

INTRODUCTION: Previously, the authors investigated the feasibility of infusing boric acid to a radiochromic gel formula made of polyvinyl alcohol (PVA), glutaraldehyde (GTA), and iodide (I); also known as the PVA-GTA-I gel dosimeter. The results of the study demonstrated that boric acid can be infused to the PVA-GTA-I formula gel to increase the sensitivity of the gel to neutron irradiations. The standard PVA-GTA-I formula has been previously reported to be capable of reuse through annealing, where the irradiated gel samples were converted from red to clear after heating with absorbance values similar to fresh and unirradiated samples [1,2]. In the present study, we aim to verify if the same capability can be achieved with boric acid-infused PVA-GTA-I gels after long-term storage of post-irradiation with thermal neutron beams.

EXPERIMENTS: All gel samples were prepared using ultrapure water and analytical grade chemicals. The standard formula of the PVA-GTA-I formula was infused with different boric acid concentrations of 0, 25, and 50 mM. The liquid mixtures were poured into PMMA cuvettes and stored inside a vacuum chamber (-0.08 MPa) for 3-hours. The cuvette samples were then covered with PE lids and stored inside a sterilizing oven at 50°C for 12 hours to convert from liquid to gel.
The neutron irradiations of the samples were performed at the Heavy Water Neutron Irradiation Facility (HWNIF) of Kyoto University Reactor (KUR) with a 1MW nominal power. The samples were fixed on a rail system and irradiated at different periods: 20-, 40-, 80-, and 120-min. After irradiation, the samples were stored at room temperature and measured 55 days after irradiation. The optical absorbances were obtained using a UV-Vis spectrometer (Thermo Fisher Scientific Inc., USA).
After the initial reading, the gel samples were annealed in the oven at 50°C at time intervals of 24, 48, 72, and 96 hours. It should be noted that after each annealing time interval, the samples were stabilized at room temperature for 2 hours before measurement.

RESULTS: Figure 1 shows the dose-response of the gel samples of different boron concentrations. It can be observed that linear dose-response can be achieved with all gel formulations despite the different sensitivities and natural oxidation in the gel after longer storage time. The annealing results for each gel set in Fig. 2 shows that boric acid-infused PVA-GTA-I gels can be annealed even after 55 days post-irradiation, albeit longer annealing time for formulations with higher boric acid concentration.
These findings highlight the possible reuse of the boric acid-infused PVA-GTA-I gels, which is a desirable property for repeated dosimetry using thermal neutron beams.

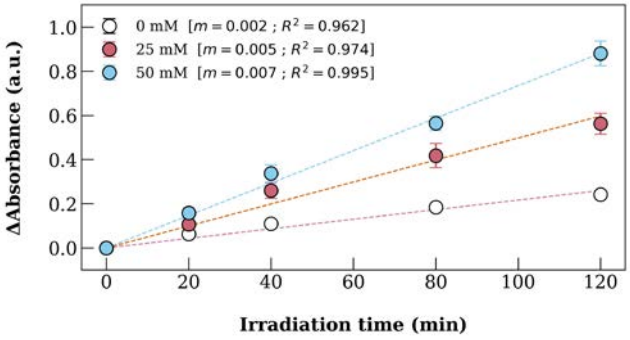

Fig. 1 Dose-response curves of PVA-GTA-I gel samples with various boric acid concentrations. All samples were measured 55 days after irradiation.

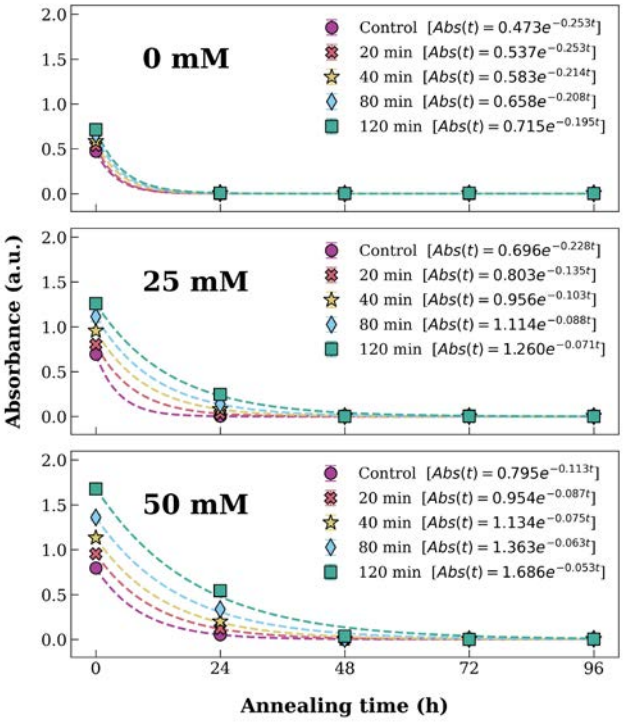

Fig. 2 Absorbance decay plots of the gel samples with different boric acid concentrations annealed at 50°C. All data sets were fitted with an exponential decay function: $A(t) = A_0 * \exp(-kt)$.

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PR9-20 Establishment of Quantitative Measurement of Boron Concentration Distribution in vivo by Imaging of Prompt Gamma Rays

S. Komura, T. Mizumoto, T. Mitani, Y. Sakurai[1], N. Kondo[1], M. Suzuki[1], T. Takata[1], H. Kimura[2], T. Tanimori[3] and A. Takada[3]

Fukushima SiC Applied Engineering Inc.
[1]Institute for Integrated Radiation and Nuclear Science, Kyoto University
[2]Department of Analytical and Bioinorganic Chemistry, Kyoto Pharmaceutical University
[3]Graduate School of Science, Kyoto University

INTRODUCTION: Tomographic imaging of 478 keV prompt gamma rays emitted from the $^{10}B(n,\alpha)^{7}Li$ reaction is required to monitor the dose distribution in a patient during boron neutron capture therapy (BNCT). Several gamma-ray imaging systems using SPECT technique have succeeded in imaging large phantoms with a high concentration of ^{10}B [1,2]. However, since their detection efficiency is significantly reduced due to their very thick collimators, they have not been successful in imaging living tumor tissues which is typically small and with a low ^{10}B concentration less than 100 ppm when BPA is administered.
To overcome the current situation, we have been developing a new gamma-ray imaging system using the Electron Tracking Compton Cameras (ETCCs) which determine the arrival direction of gamma rays event by event using the electric collimation. In our previous studies, we demonstrated that our ETCC provides high contrast images of 478 keV prompt gamma rays emitted from a large water phantom with a ^{10}B concentration of 1000 ppm using E-3 Neutron Guide Tube (E-3) at Kyoto University Reactor (KUR) at the 1-MW operation [3].

EXPERIMENTS: We carried out prompt gamma-ray imaging studies using four U87MG tumor-bearing mice treated with BPA. Each mouse was anesthetized and administered 1000 mg/kg of BPA by tail vein injection, then fixed in a plastic case and irradiated with E-3 thermal neutron beam for 90 minutes at the 5-MW operation, one mouse at a time. The case was placed just above the ETCC. The ETCC measured the 478 keV prompt gamma rays emitted from the mouse during the irradiation time and reconstructed a back-projection image.

RESULTS: We were the first to successfully obtain 478 keV prompt gamma-ray images of BPA-treated tumor-bearing mice. For patent reasons, the details are not presented here.

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