NEUTRON SPECTROMETRY OF THERMAL COLUMN BY VARIOUS FILTER/MODERATING MATERIALS AT REACTOR TRIGA PUSPATI FOR BNCT RESEARCH

Mohd Rafi Mohd Solleh[1], Abdul Aziz Mohamed[2], Abd. Aziz Tajuddin[3], Faridah Idris[4], Megat Harun Ar-Rashid, Mohamad Hairie Rabir[4], Muhammad Rawi Md Zin[4], Hafizal Yazid[4], Azraf Azman[4]

[1]*Fakulti Sains dan Bioteknologi, Universiti Selangor, 45600 Bestari Jaya, Selangor Darul Ehsan, Malaysia.*

[2]*Centre for Nuclear Energy, College of Engineering, Universiti Tenaga Nasional, 43000 Kajang, Selangor, Malaysia.*

[3]*Pusat Pengajian Sains Fizik, Universiti Sains Malaysia, 11800 Penang, Malaysia.*

[4]*Agensi Nuklear Malaysia, 43000 Kajang, Selangor, Malaysia.*

ABSTRACT

Thermal neutron beam from thermal column was selected for a Boron Neutron Capture Therapy (BNCT) system utilizing the Malaysian TRIGA MARK II reactor. Determination of shielding materials for fast and epithermal neutron was conducted. The materials selected were polyethylene, paraffin and water. For gamma-ray shielding, lead was used. The objective of this paper is to present the simulation and verification of an optimal design of BNCT collimation at a beam line viewing the thermal column. A collimator was made from polyethylene pipe with 8 cm of diameter filled with paraffin.

ABSTRAK

Neutron terma merupakan neutron yang serasi untuk sistem 'Boron Neutron Capture Therapy' (BNCT) yang menggunakan reaktor TRIGA Malaysia MARK II. Penentuan dan bahan-bahan untuk neutron puasa dan epithermal telah dijalankan. Bahan-bahan yang dipilih ialah polietilena, dihadiri oleh penyelidik bagi dan air. Bagi melindungi sinar gamma, plumbum yang digunakan. Objektif kertas kerja ini adalah untuk membentangkan simulasi dan pengesahan Reka bentuk BNCT collimation yang optimum pada garisan pancaran yang melihat tiang haba. Collimator yang telah dibuat dari paip polietilena dengan 8 cm diameter yang dipenuhi dengan dihadiri oleh penyelidik bagi.Keywords: composite, thermography, rad-waste, impact strength.

INTRODUCTION

A very promising cancer treatment called Boron Neutron Capture Therapy (BNCT) is selected to be studied in this research. BNCT is a radiation therapy for the treatment of cancers like melanoma and glioblastoma multiforme (Yanagie et al., 2010). BNCT is done by firstly, a stable isotope of boron-10 (10B) is administered to the patient via a carrier drug and then the patient is irradiated with a

neutron beam. The ^{10}B will then undergo the capture reaction $^{10}B(n, \alpha)^{7}Li$ where 10B capture cross section for thermal neutrons is 3840 barn (Valda et al., 2005). This is the scientific fact on why thermal neutron is used in BNCT.

Neutron spectrum that came out from the hole in thermal column door is suitable for BNCT. The highest count is around low or thermal neutron energy <0.5 eV (IAEA, 2001) and was decreases as neutron energy increases. This spectrum profile is as expected and as needed for BNCT research. It gives a very good start for BNCT research in Malaysia.

In building BNCT facility with thermal neutrons, few spectra of epithermal neutrons can be moderated to thermal neutrons and this will increase the intensity of the available thermal neutrons. According to IAEA (2001), an epithermal neutron energy range is 0.5 eV to 10 keV. Fast neutron spectrum here was very low so no additional shielding is needed. In BNCT, the energy range is taken as >10 keV for fast neutrons. Fast neutrons, which invariably accompany the incident beam, have a number of undesirable characteristics such as the production of high LET protons with a resulting energy dependence of their induced biological effects. Therefore, as much as possible it is one of the main objectives of BNCT beam design to reduce the fast neutron component of the incident beam.

This can be done by using some materials. Cadmium produced the highest thermal neutron peak followed by B4C. Surprisingly, lead can also be used beside either ice or water.

MATERIALS AND METHOD

Neutron Flux measurement

Neutron flux measurements were done with Fission Chamber detector (see Figure 1). Back wall of the shielding box had a hole on it with 10.0 cm of diameter. A small polyethylene collimator was placed inside the shielding box to reduce the beam size for safety precautions. Fission Chamber detector was then placed at the back of the shielding box. Water tank was placed at the back of the Fission Chamber detector as biological shielding. Neutron flux measurement was done at the reactor power of 100 kW for 1 hour.

Fission Chamber has two coaxial electrodes. The inter-electrode space is filled with pressurized gas (e.g. argon at 1.5 bar). A thin layer of fissile material (from a few mg to a few hundreds of mg) is deposited on the inner electrode (namely the anode). It is to induce a fission that generates two heavily charged ions, and the fission fragments, emitted in two nearly opposite directions immediately after a neutron reaches the fissile deposit. The one emitted out of the deposit ionizes the filling gas on its trajectory (Filliatre et. al., 2008). The electrons and positive ions are separated and drift across the gas, generating a current signal that can be amplified and processed when a direct current (DC) voltage of a few hundred volts is applied between the electrodes. That DC voltage must be low enough to prevent the production of secondary ionization pairs (a phenomenon occurring in proportional counters) and high enough to collect all the charges. The fission chamber works in the saturation regime, namely for which the neutron-induced current signal is proportional to the fission rate if both conditions are fulfilled (Filliatre et. al., 2008).

Under irradiation, high energy fission products (about 90 MeV per each on average for 235U) are emitted in opposite directions when neutrons induce fission reactions inside the deposit. Thus, one is absorbed in the anode while the second crosses the inter-electrode space, ionizing the filling gas on its path and consequently generating a high number of electron–ion pairs. An electric field is then generated between the two electrodes, involving a migration of charges when a voltage is applied. The collected charges are responsible for the creation of an electric current. The layout of this current according to the voltage applied gives a characteristic curve, known as the calibration curve (Chabod et. al., 2006).

Fission Chamber can be operated in three modes, namely pulse, campbelling and current modes, depending on the fission rate within its fissile deposit and the measurement system. A fission chamber is an ionization

chamber with a fissile deposit in order to be sensitive to neutrons (Loiseau et al., 2013). Jammes et al. (2012) had reported that the signal produced by fission chamber detector and detected by their PHI4 detector with a coating thickness of about 100 mgcm-2 resulted a loss of less than 1%.

In monitoring and controlling the neutron flux, Fission Chambers have been largely used in fission reactors. The behaviour of a Fission Chamber is determined by its calibration curve (measured current versus polarization voltage). Uncertainties for Fission Chamber is below 5% for fission rates up to fluences of 1022 ncm-2 (Cabellos et al., 2010). Charged particles crossed the argon atmosphere between the two electrodes of the chamber (anode and cathode) and ionized the gas on their trajectory creating an average of 0.3 ion pairs per nm (Ar+, e-). The fission reaction turns an uncharged particle (neutron) into charged particles (Fission Fragment, FF) which are easier to detect (Poujade and Lebrun, 1999):

$$(\mathrm{n}, {}^{235}\mathrm{U}) \rightarrow (2\mathrm{FF}, 2\text{- }3 \text{ neutrons, energy}) \quad (1)$$

where one fission reaction gives birth to two fission fragments. Their average energy is around 90 MeV for the lightest and 60 MeV for the heaviest.

Fission Chamber is designed to function in high gamma-ray radiation fields and does not need massive lead shielding. To have an accurate neutron count rate, pulse amplitude discrimination is used to eliminate the unwanted pulses due to alpha decay of uranium and gamma radiation. The energy of uranium fission fragments, originating from thermal neutrons interactions, is typically 165 MeV. The energy of the alpha particles from uranium decay is typically 4.7 MeV, while gamma-ray photons deposit even less energy per interaction (Mohindra et al., 2007).

Neutron Spectrum Measurement

Before the neutron spectrum measurement with MICROSPEC-2 spectrometer (Figure 2), the suitability and safety in using MICROSPEC-2 spectrometer was analysed in an experiment as in Figure 3. This experiment was conducted after the reactor was not operated for two days (reactor is shut down) to avoid unnecessary radiation exposure of radiations produced from reactor operation. Firstly, a crane was used to take out the concrete plug from thermal column door. The concrete plug then was replaced with a collimator. The collimator was made from paraffin that filled in polyethylene pipe with 8 cm of holes diameter.

After the collimator was ready, a shielding box was placed on thermal column door to shield the hole. There is a second small collimator made from polyethylene placed inside the shielding box with a hole of 1.5 cm diameter. This second collimator was used to reduce the neutron beam diameter for safety reason and to prevent the spectrometer from damage because MICROSPEC-2 is built for measuring a low neutron flux only.

After MICROSPEC-2 was prove to be safe to measure neutron beam at reactor power of 100 kW, neutron measurement was continued by using a same shielding box but the rear shielding wall was changed with a new one. This new wall had a hole of 10 cm in diameter at its centre. All set up is remain the same. MICROSPEC-2 spectrometer was placed at the back of the shielding box. Water tank then was placed after the spectrometer. It is used as a biological shielding. Neutron spectrum measurement was done with MICROSPEC-2 spectrometer at reactor power of 100 kW for 1.9 hours.

After the measurement was completed, a 0.1 cm thick cadmium plate was placed covering the hole on the shielding box and before MICROSPEC-2 spectrometer. Neutron spectrum measurement at 100 kW of reactor power was repeated with MICROSPEC-2 spectrometer for 1.0 hour. This measurement was mean to find the best moderator to be used to moderate fast and epithermal neutron to thermal neutron in BNCT treatment at Malaysian TRIGA MARK II reactor.

Cadmium plate was then replaced with 0.4 cm thick B_4C plate. Neutron spectrum measurement at 100 kW of reactor power was repeated with MICROSPEC-2 spectrometer for 1.0 hour. B_4C plate was then replaced with lead plate of 0.6 cm thick. Neutron spectrum measurement at 100 kW of reactor power was done with MICROSPEC-2 spectrometer for 35.0 minutes. Lead plate was then replaced with water in an aluminium case

of 10.0 cm height, 8.0 cm width and 10.0 cm length. Neutron spectrum measurement at 100 kW of reactor power was done with MICROSPEC-2 spectrometer for 25.3 minutes. Water was then replaced with ice cube in the same aluminium case. Neutron spectrum measurement at 100 kW of reactor power is done with MICROSPEC-2 spectrometer for 28.6 minutes. The period of each measurement is not equal because the periods have to be made shorter for the last part of experiments to accommodate the time allocated for these experiments.

Figure 1. Neutron Fission Chamber detector

Figure 2. Neutron spectrometry detector

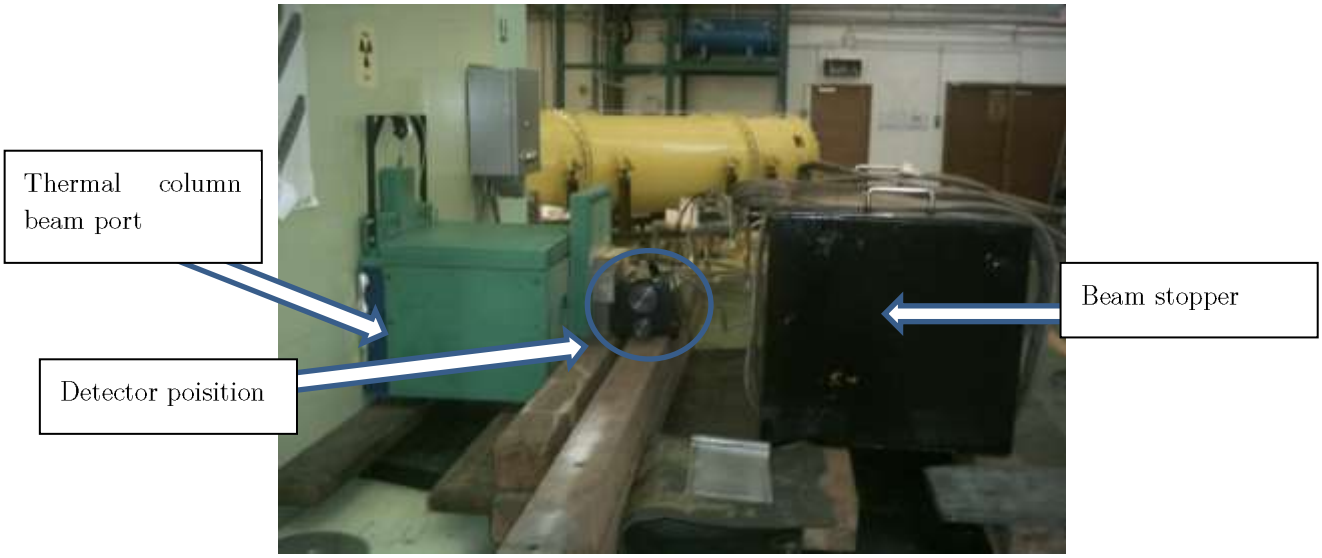

Figure 3. Detection Set-up

RESULTS AND DISCUSSION

Neutron beam that came out from the hole of thermal column door was measured with Fission Chamber detector. Table 1 shows the counts of neutron particles measured with the Fission Chamber detector. The measurement was done in a period of one hour and at reactor power of 100 kW. The average of neutron count per second was 2.30 x 10^4 s^{-1}. Neutron flux calculated from the neutron count for exposed area of 1.8 cm^2 on the Fission Chamber detector was 1.35 x 10^4 $cm^{-2}s^{-1}$. Calculated neutron flux for 1 MW of reactor power was 1.35 x 10^5 $cm^{-2}s^{-1}$.

Table 1. Counts of neutron particles measured with the Fission Chamber detector.

Time in Seconds	600	1200	1800	2400	3000	3600	Average
Neutron per Seconds ($\times 10^4$)	2.30	2.40	2.40	2.40	2.40	2.20	2.30

Figure 4 shows the profiles of materials tested using the neutron spectrometer equipment (Micro-spec). Ice has shown the highest thermal neutron count rate.

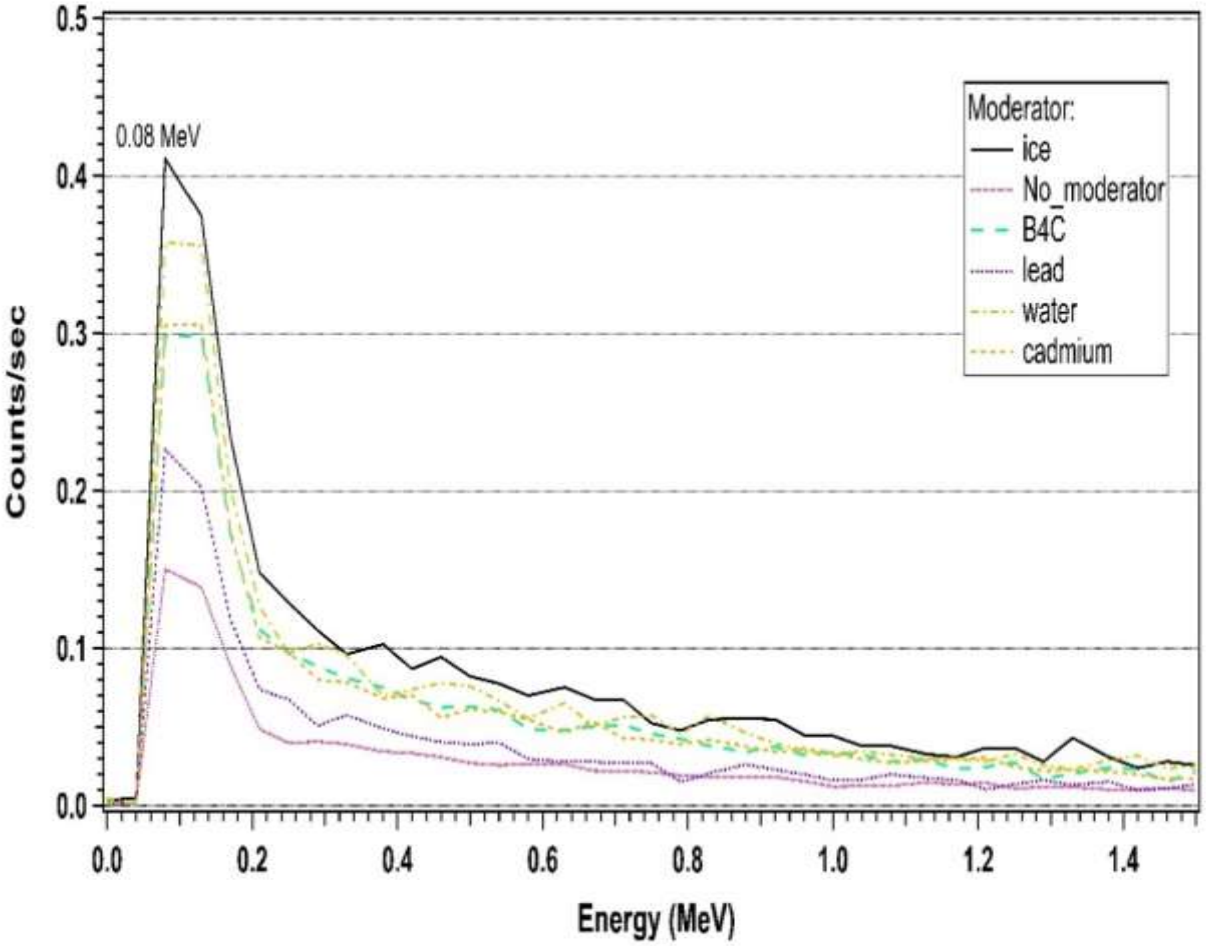

Figure 4. Combination of all neutron spectrums thermal column beam port with and without filter/moderator materials

CONCLUSION

The fluxes measured were very much lower than all measurements done with TLD detectors. This was believed because of the aperture used in this measurement that only allowed a very small neutron beam (1.5 cm of beam diameter). The measurement with TLDs use neutron beam with 8 cm of diameter. A bigger aperture will give higher flux and will be sufficient to be used for BNCT treatment. Neutron spectrometry for ice, B_4C, lead, water and cadmium has been established.

ACKNOWLEDGEMENT

Special thanks to Composite Technology Malaysia for providing the test materials. Our gratitude go also to Universiti Tenaga Nasional, Universiti Sains Malaysia, Universiti Selangor, Hokkaido University and Nuclear Malaysia for supporting this work.

REFERENCES

Cabellos, O., Ferna´ndez, P., Rapisarda, D., and Garcı´a-Herranz, N. (2010). Assessment of fissionable material behaviour in fission chambers. Nuclear Instruments and Methods in Physics Research A. Vol. 618, pp.248–259.

Chabod, S., Fioni, G., Letourneau, A., and Marie, F. (2006). Modelling of fission chambers in current mode—analytical approach. Nuclear Instruments and Methods in Physics Research A. Vol. 566, pp.633–653.

Filliatre, P., Oriol, L., Jammes, C., and Vermeeren, L. (2008). Reasons why Plutonium-242 is the best fission chamber deposit to monitor the fast component of a high neutron flux. Nuclear Instruments and Methods in Physics Research A. Vol. 593, pp.510– 518.

IAEA (2001). Current Status of Neutron Capture Therapy. IAEA-TECDOC-1223.

Jammes, C., Filliatre, P., Loiseau, P., and Geslot, B. (2012). On the impact of the fissile coating on the fission chamber signal. Nuclear Instruments and Methods in Physics Research A. Vol. 681, pp.101–109.

Loiseau, P., Geslot, B., and Andre, J. (2013). On the fission chamber pulse charge acquisition and interpretation at MINERVE. Nuclear Instruments and Methods in Physics Research A. Vol. 707, pp.58–63.

Mohindra, V. K., Vartolomei, M. V., and McDonald, A. (2007). Fission chambers for CANDU® SDS neutronic trip applications. Proceedings of the 28th Annual Canadian Nuclear Society (CNS) Conference, Saint John, New Brunswick, Canada. Pp. 787.

Poujade, O., and Lebrun, A. (1999). Modeling of the saturation current of a fission chamber taking into account the distorsion of electric field due to space charge effects. Nuclear Instruments and Methods in Physics Research A. Vol. 433 pp.673-682.

Valda A., Minsky D. M., Kreiner A. J., Burlon A. A. and Somacal H. (2005). Development of a tomographic system for online dose measurements in BNCT (Boron Neutron Capture Therapy). Brazilian Journal of Physics, Volume 35, no. 3B. Pp.785-788.

Yanagie H., Kumada H., Nakamura T., Higashi S., Ikushima I., Morishita Y., Shinohara A., Mitsuteru F., Suzuki M., Sakurai Y., Sugiyama H., Kajiyama T., Nishimura R., Ono K., Jun N., Minoru O., Eriguchi M., and Takahashi H. (2010). Feasibility evaluation of neutron capture therapy for hepatocellular carcinoma using selective enhancement of boron accumulation in tumour with intra-arterial administration of boron-entrapped water-in-oil-in-water emulsion. Appl Radiat Isot. Volume 69(12), pp.1854-1857.