IOP Conf. Series: Materials Science and Engineering **736** (2020) 062023 doi:10.1088/1757-899X/736/6/062023

A design on the collimator for boron neutron capture therapy (BNCT) research facility at the thermal column of TRIGA MARK II

S Shalbi[1], N Sazali[1,2]* and W N Wan Salleh[3]

[1]Faculty of Mechanical Engineering, Universiti Malaysia Pahang, 26600 Pekan, Pahang, Malaysia
[2]Centre of Excellence for Advanced Research in Fluid Flow (CARIFF), Universiti Malaysia Pahang, Lebuhraya Tun Razak, 26300 Gambang, Kuantan, Pahang, Malaysia
[3]Advanced Membrane Technology Research Centre (AMTEC), School of Chemical and Energy, Faculty of Engineering, Universiti Teknologi Malaysia, 81310 Skudai, Johor Darul Takzim, Malaysia

*azlianie@ump.edu.my

Abstract. The development of the Boron Neutron Capture Therapy (BNCT) facility in Malaysia can be performed at the thermal column of the Malaysia research reactor. TRIGA MARK II is one of the facilities that can provide neutron source for BNCT facility. The specification of neutron flux and the gamma dose rate must consider for the development of the BNCT facility as a safety precaution for this research. Based on previous research, the thermal column identified as a suitable place for BNCT facility. To design the neutron collimator for BNCT purpose, the characterization of material towards thermal neutron flux explored using TLD and Microspec-6 and the collimator design was simulate using Monte Carlo N-Particle (MCNP) software based on the characterize materials in order to produce high thermal neutron flux. The combination of lead, HDPE, 30% borated polyethylene and aluminium as collimator design D1 simulate the highest thermal neutron 1.5770×10^9 neutron.cm^{-2}s^{-1} and suitable for BNCT research purpose at the thermal column.

1. Characterization of material for BNCT collimator

The nuclear application, such as BNCT required explicit material to avoid any radiation leaking for safety purpose. Therefore, to design the suitable collimator for BNCT, the characteristic of the material towards neutron flux must be studied. The characterization of material for BNCT collimator done by an experimental approach (TLD and Microspec-6) at the thermal column. The characterization of material was essential to ensure the material used for design the collimator can optimize the thermal neutron beam and reduces other ionizing particles. The collimator for BNCT facility required high thermal neutron compared to the epithermal neutron and fast neutron [1]. The material selected based on the previous study, as summarized in table 1,2 and 3. Table 1 shows the list of neutrons shielding material and the thickness required to shield neutron and photon for BNCT by a few researchers. Table 2 shows the neutron collimator material studied by a few researchers for BNCT purpose, while table 3 shows a list of neutron moderator material used by a few researchers for BNCT. All of the materials list from the table was useful in this research to design and modify the collimator for BNCT research facility. However, in this research, only a few materials were used due to the availability and cost of the material.

IOP Conf. Series: Materials Science and Engineering **736** (2020) 062023 doi:10.1088/1757-899X/736/6/062023

Table 1. Neutron shielding material studied by a few researchers for BNCT purpose.

Material	Researchers
Bismuth	[2]
Fluental	[3]
Aluminum, Cadmium, Fluental, and Bismuth	[4]
Difluorocarbene	[5]
Calcium fluoride	[6]
Heavy Water	[7]
Cadmium	[8]

Table 2. Neutron collimator material studied by a few researchers for BNCT purpose.

Material	Researchers
Lithium	[9]
Polyethylene	
Polyethylene	[10]
High Density Polyethylene (HDPE)	[6]
Lead	[5]
Paraffin	[8]

Table 3. Neutron moderator material studied by a few researchers for BNCT purpose.

Material	Researchers
Aluminium	[2]
Heavy Concrete	[3]
Ordinary concrete, and 5%, and 30% borated Polyethylene	[11]
Polyethylene	[8]

1.1. Thermoluminescent dosimeter

This research is conducted using two thermoluminescent dosimeters (TLD) which are TLD-600 and TLD-700 that have commonly used around the world as personal dosimetry. The name of TLD-600 and TLD-700 based on the material used as a detector. TLD-700 made from lithium-7 isotope fluoride with adequate atomic number (Z) for photoelectric absorption of 8.2. Besides that, TLD-700 have about 35000-6000 ranges of emission spectra equally with TLD-600 that made from the lithium-6 isotopes fluoride with 8.2 atomic number (Z). The main difference between TLD-600 and TLD-700 is the composition of Li-6 in TLD-700 is 0.0007% and 95 for TLD-600. Meanwhile, the composition of Li-7 in TLD-700 is 99% while only 4.38 for TLD-600. The main reason for using TLD for the research because of the characteristic of TLD towards the radiation itself. TLD was well known with the capability for a long term of data storage and highest sensitivity towards radiation. The different types of TLD have a different function, and for example, the TLD-700 can only measure photon while the TLD-600 are sensitive with both photon and neutron [1]. Moreover, the element of LiF inside the TLD detector is similar enough with the human tissue in terms of atomic density which make use of TLD is convenient. Both TLD-600 and TLD-700 are determined to have practically been equipped to detect both thermal neutron and gamma-ray. TLD-700 dosimeter was less sensitive towards the thermal neutron because of the low probability of Li capture the thermal neutron reaction [12].

1.2. Microspec- 6 N probe spectrometer

In this research, the Microspec-6 N probe spectrometer has been used to measure the neutron spectrum at the BNCT facilities during the experimental work and to study the characteristic of material towards

IOP Conf. Series: Materials Science and Engineering **736** (2020) 062023 doi:10.1088/1757-899X/736/6/062023

neutron and gamma. Basically, the measurement of neutron energy is to measure the neutron dose in order to maximize the radiation shielding [13]. The Microspec-6 was built to measure a low neutron flux which is suitable to measure the neutron energy spectra that expect to have higher thermal neutron (slow neutron) flux for BNCT purpose. In order to get sufficient result, the experiment must be conducted after two or more days the reactor being shutdown to avoid unnecessary radiation that can damage the Microspec-6 and get poor result with the dead time more than 25% (which is consider as unaccepted result). Microspec-6 spectrometer can be used not only to measure neutron energy spectrum, but can be used to measure gamma, x-ray and beta energy spectrum as well based on the probe use. As this research was considering neutron as a main focus, the N-probe (neutron probe) has been used. In general, Microspec-6 spectrometer was recognized as dominant portable spectroscopy system as the capabilities to handle widely task such as measurement, detection, dosimetry, identification of ionizing radiation and the radiation mapping. Figure 1 shows the Microspec-6 N-probe spectrometer.

Figure 1. The Microspec-6N-probe spectrometer [4].

From the figure 1, the Microspec-6 N-probe spectrometers have two separate detectors. Both detectors are functional to detect the neutron with the energy range from 0.025 keV to 20 MeV. The neutron with the energy ranges from 0.025 keV and 800 keV are done using ^{3}He proportional counter. Meanwhile, the neutrons with an energy range from 800 keV to 20 MeV are determined using liquid scintillator [14]. Figure 2 shows the setup of the Microspec-6N-probe spectrometer in the material characterization experiment carried out at the thermal column in this research.

IOP Conf. Series: Materials Science and Engineering **736** (2020) 062023 doi:10.1088/1757-899X/736/6/062023

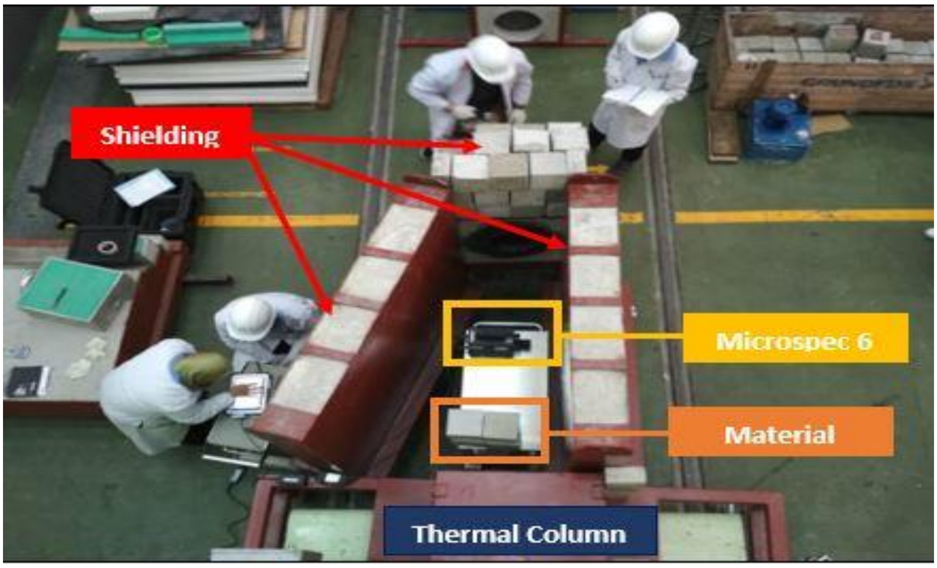

Figure 2. The Microspec-6N-probe spectrometer setup for material characterization experiment.

1.3. Background radiation measurement

The background radiation measurement important as a safety precaution during handling the experiment in this research. The background of gamma dose and neutron dose must be measure to avoid external radiation exposure and ensure the experiment was safely performed. The neutron background radiation dose is done by fixed the detector such film badge or TLD with placed on the suitable area for the suitable period time at the research reactor hall. In order to measure the gamma dose, the portable survey meter is used in three phase which is Phase 1 (pre-experiment), Phase 2 (during the experiment) and Phase 3 (post experiment). In Phase 1, the background radiation was measured before the experiment when the research reactor did not operate for at least overnight. This is to ensure, there are no abnormal exposure for the safety purpose. After that, in the Phase 2, the background radiation was measured during the ongoing experiment. The Phase 2 was the most important phase to measure background radiation because the research reactor is operated thus the radiation dose at the surrounding will increase linearly. As a safety precaution, the background radiation in Phase 2 was measured all the time to ensure the radiation exposure does not exceed safety limits.in the Phase 3 (post-experiment), the portable survey meter was used to measure the background radiation after the experiment and the research reactor being shut down. This procedure is required to maintain the radiation dose at the safety level of the reactor. Figure 3 shows the portable survey meter.

Figure 3. Portable survey meter.

IOP Conf. Series: Materials Science and Engineering **736** (2020) 062023 doi:10.1088/1757-899X/736/6/062023

2. Characterization of material for collimator design

The collimator design must consider three types of material purpose which are moderator material, shielding material and neutron collimator material. In this research, the material such as Polyethylene, Paraffin, HDPE (High Density Polyethylene), Lead, Cadmium, 30% Borated Polyethylene, and 5% Borated Polyethylene was used based on the previous study and the availability and the cost.

2.1. Characterization of material using Microspec-6

The study of a characteristic of material using Microspec-6 is important as to ensure the material purpose and characteristic towards the neutron. The neutron spectrum profiles are recorded using Mobile Microspec software which directly counts from neutron probe. Neutron release from the thermal column is used as the neutron source for this experiment. The use of the thermal neutron as a neutron source can validate the characteristic and the criterion for the preliminary collimator material composition and arrangement for BNCT. Material sample specification for this experiment was 5 cm for thickness, 10 cm for width and 25 cm for height for all material.

2.1.1. Neutron spectrum profiles. The neutron spectrum profiles from the open source were recorded as shown in figure 4 at the optimal reactor power TRIGA MARK with 100 kW. The intensity of neutron delivered to the Microspec-6 from open source is counted starting from 2.20 x 10^4 n/cm^2 and was decrease as the energy increase.

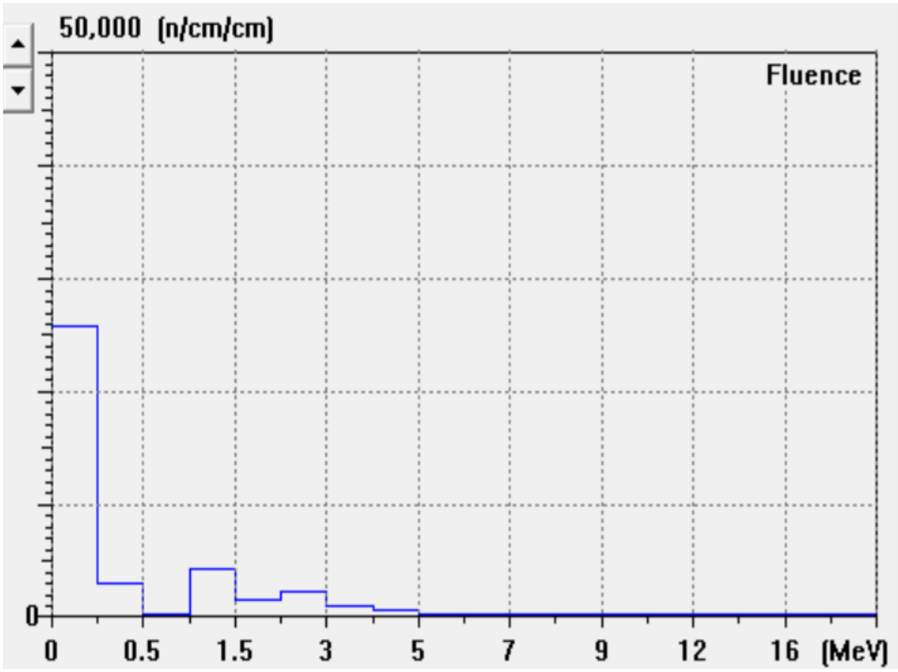

Figure 4. Neutron spectrum profile obtained from the open beam source.

From the figure 4 show that the open source of neutron source has high thermal or low neutron energy as expected and was decreasing as the neutron [15]. Even though there are some disturbance and noise which lead to little error in figure 4, but the average pattern shows the same trend line which can be used for this experiment analysis. All the neutron spectrum profile for other samples are shown in figure 5.

IOP Conf. Series: Materials Science and Engineering **736** (2020) 062023 doi:10.1088/1757-899X/736/6/062023

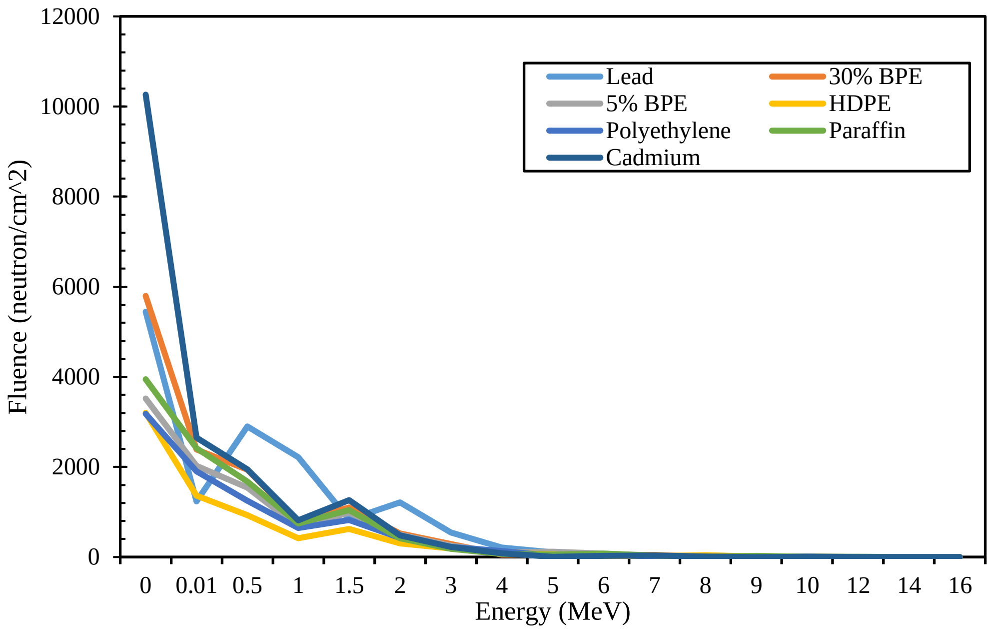

Figure 5. Neutron spectrum profile obtained from Microspec-6 detector for paraffin, HDPE, polyethylene, 5% borated polyethylene, 30% borated polyethylene, lead and cadmium.

Based on the figure 5, all the material shows the same trend line with higher fluence on the thermal neutron energy region except for lead material that have higher fluence on epithermal neutron energy region and fast neutron energy region. Most of the material have high fluence at the thermal neutron energy region because of the neutron source coming from the thermal column mostly was thermal neutron. This show that the neutron source from the thermal column was reliable to be used for BNCT purpose. The graphical in figure 5 also show that cadmium has the highest fluence compare to others material with more than 10,000 neutron/cm^2 on the region of thermal neutron and followed by lead and 30% BPE with the fluence range of 5,000 and 6,000 neutron/cm^2 and others material obtained less than 4,000 neutron/cm^2. Based on the entire material sample tested, the arrangement of the thermal peak of the neutron spectrum profile is tabulated in table 4. From the table 1, this research can observe the different of each sample thermal peaks.

Table 4. Thermal peaks recorded.

Material	Thermal Neutron Peak(neutron/cm^2)
Paraffin	5.81×10^3
Polyethylene	4.88×10^3
Lead	8.90×10^3
Cadmium	1.25×10^2
30 % Borated Polyethylene	8.28×10^3
5% Borated Polyethylene	5.87×10^3
High Density Polyethylene	4.52×10^3

All samples measured the decrement of the thermal neutron peaks when irradiated with the neutron source. This proved the readable signal of neutron source use for this measurement was relevant and healthy. In correlation, the RTP have efficient and sufficient neutron source for BNCT research. The

IOP Conf. Series: Materials Science and Engineering **736** (2020) 062023 doi:10.1088/1757-899X/736/6/062023

thermal neutron peaks show the tendency of each material to moderate, absorb and shield the thermal neutron from the neutron source of RTP. The lowest thermal neutron peaks recorded show that the material have strong absorption of thermal neutron characteristic. Otherwise, the higher thermal neutron peak was the types of materials were suitable to be proposed as neutron collimator material for BNCT facility at thermal column. Based on **t**able 4, the highest thermal neutron peaks were recorded by lead with 8.90 x 10^3 neutron/cm^2 then followed by 30% borated polyethylene and the least was measured by cadmium with 1.25 x 10^2 neutron/cm^2.

2.1.2. Neutron and gamma attenuation coefficient. The effective shielding material for collimator can be determined based on the gamma attenuation coefficient calculated from each irradiated material sample based with the gradual decrease in gamma intensity against the thickness of the material sample. Distinctly, the neutron attenuation coefficient is the probability of the radiation interaction with matter per unit length and depend on the intensity of the incident neutron [8]. In consideration of material for BNCT collimator, the effective shielding material was crucial to reduce the annual accumulated dose of research and radiation worker involved in this BNCT research. Generally, the neutron intensity by shielding material was directly proportional to the gamma ray. Hypothesis stated that the lower neutron intensity along the material thickness means the lower gamma produced [1]. The formulation of neutron attenuation coefficient can be calculated. The gamma and fast neutron attenuation coefficient data has been tabulated on the figure 6 in term of the percentage of reduction of gamma and fast neutron on the material sample. A fast neutron can be converted into the slow neutron based on the material use while gamma was negotiable for BNCT purpose. Thus, the attenuation coefficient for fast neutron and gamma are important to identify the material used for moderate and shielding neutron and material that can be used to shield gamma for the collimator design. As expected, the lead slab resulted in high gamma absorption. As the results, lead are widely use in radiation therapy room as a gamma shielding. However, the lead has the least percentage of fast neutron reduction but the high-density polyethylene has the highest percentage with 70% of the fast neutron being absorbing during the measurement. Based on this finding, the characteristic of material towards fast neutron and gamma ray was observed as lead have highest efficiency in shielding gamma ray while HDPE was effective in fast neutron shielding. Those materials were important to use in designing the BNCT collimator.

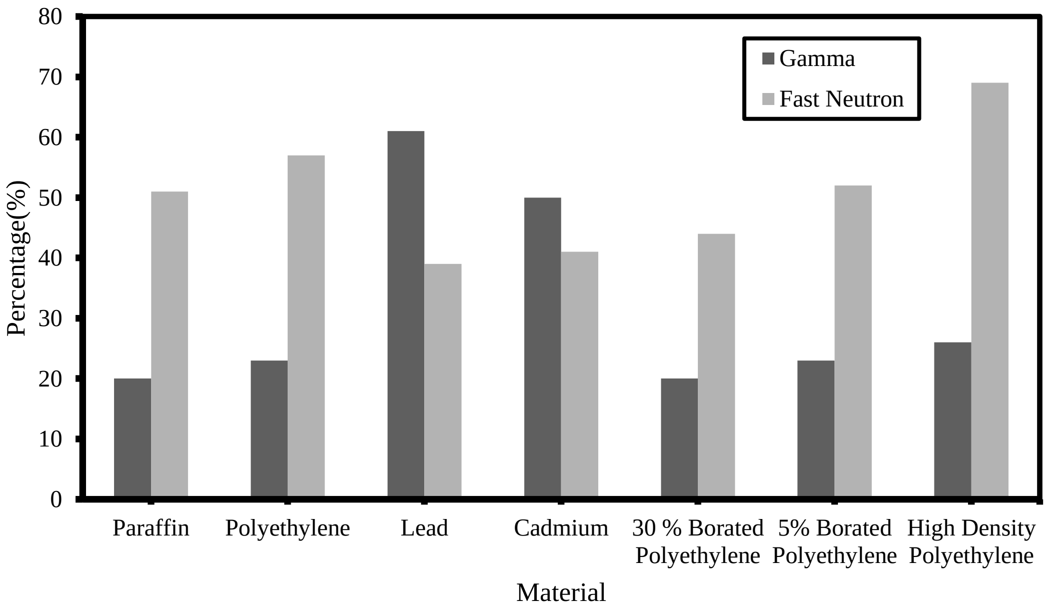

Figure 6. The percentage of gamma and fast neutron shield by material based on linear attenuation coefficient.

Beside of the shielding characteristic, the neutron attenuation coefficient also has been used in order to identify the moderating material. The optimization of beam for BNCT purpose required maximal thermal neutron and minimal gamma and another beam. According to Da Silva *et al.* [16], the material which can be used as a neutron moderator should have considerable scattering cross section and reduced cross section for absorption. Unfortunately, there are no such material consist of all those characteristics. Based on the table 5, the neutron attenuation coefficient for HDPE and polyethylene resulted as the highest compared to other materials with 0.39 m^{-1}. The result display that both HDPE and polyethylene have high characteristic as a moderating beam compare to other materials. This result than followed by paraffin, 5% borated polyethylene, lead, 30% borated polyethylene and lastly cadmium. The experiment result stated that the higher the density and purity of the polyethylene, the higher moderation effect towards neutron. On other side, cadmium was observed to have behavior as thermal neutron absorber rather than moderating due to the cross-section properties.

Table 5. Neutron intensity of each material with neutron attenuation coefficient.

Material	Neutron Intensity, Id (neutron/cm^{-2})	Attenuation coefficient μ(m^{-1})
Paraffin	3.52×10^3	0.38
Polyethylene	3.18×10^3	0.39
Lead	5.44×10^3	0.28
Cadmium	1.03×10^4	0.15
30 % Borated Polyethylene	5.80×10^3	0.26
5% Borated Polyethylene	3.95×10^3	0.34
High Density Polyethylene	3.20×10^3	0.39

2.2. *Thermoluminescence detector (TLD)*

After analysing the characteristic of material using Microspec-6 detector measurement. The neutron and gamma dose were measured using TLD 600 and TLD 700 detectors. The data measured was recorded in table 6.

Table 6. The TLD measurement of gamma dose and neutron dose.

Material	Neutron Dose (mSv)	Gamma Dose (mSv)
Paraffin	0.29	0.49
Polyethylene	0.21	0.46
Lead	0.096	0.15
Cadmium	0.20	0.28
30 % Borated Polyethylene	0.23	0.53
5% Borated Polyethylene	0.25	0.54
High-Density Polyethylene	0.19	0.42

At 100 kW of reactor power, the sample was put inside the shielding box and the TLD detector was positioned at the back of sample material. The result display in table 4.10 shows that lead has high absorption of neutron and gamma dose with the reduction for almost 68% from the background and 75 % for gamma dose rate. In contrast, the neutron dose counts from the paraffin superior to the other materials with 0.29 mSv while the 5% borated polyethylene resulted in higher gamma dose with 0.54 mSv. In order to construct the valid result for collimator material selection, the measurement using Microspec-6 and TLD 600 and 700 to ensure the valid material to be used for BNCT collimator design. All the result displays the characteristic for each material sample in terms of neutron collimate behaviour, thermal neutron peak, gamma and neutron shielding and the moderator behaviour towards neutron. The data from both detectors was normalized into the percentage of neutron and gamma reduce as shown in figure 7 for neutron comparison and figure 8 for gamma.

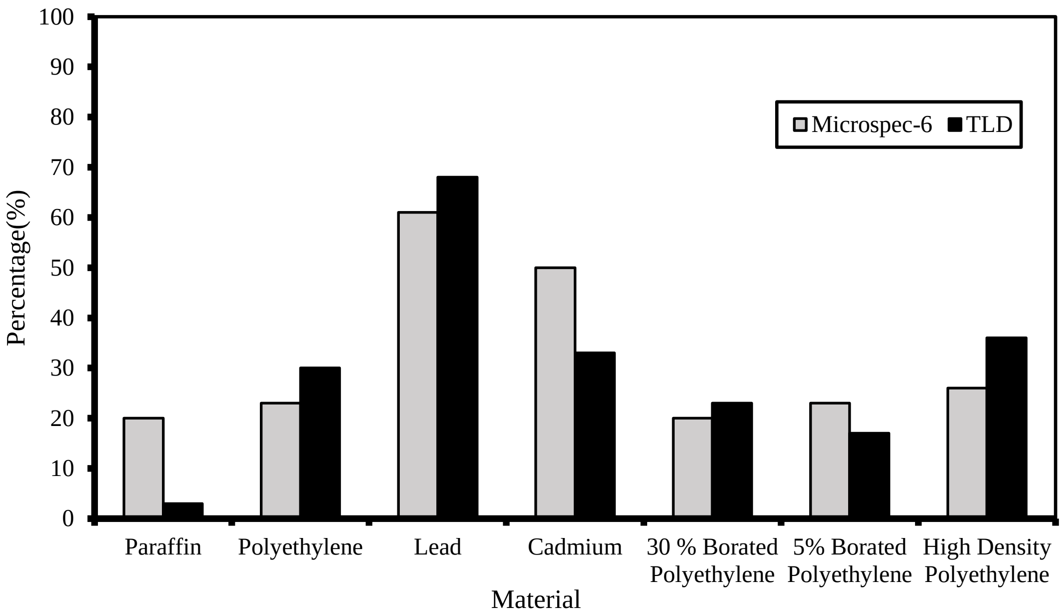

Figure 7. The percentage of neutron produce by using Microspec-6 and TLD detector.

Based on figure 7 the result shows that the paraffin has strong moderation and collimate the thermal neutron based on the percentage of neutron produce from both detectors although there was the higher difference of percentage by both detectors with 17%. Previously [8], the research concluded that the material paraffin can be used as a neutron moderator and collimate for BNCT based on the research by using Microspec-2. In terms of gamma based on figure 8, a side of paraffin, the 30% borated polyethylene, polyethylene and 5% borated polyethylene show the similar characteristic as paraffin between Microspec-6 and TLD detector for neutron and gamma. In contrast, the sample of lead, cadmium and high-density polyethylene for gamma and neutron have the same characteristic as a shielding. Based on figure 6 and figure 7, lead have a high percentage of shielding compared to other materials. This was followed by cadmium and high-density polyethylene. Between that material, only cadmium display the higher percentage different between Microspec-6 and TLD with 23%. The high percentage different of cadmium due to the functionality of Microspec-6 and TLD. The neutron spectrum from the Microspec-6 was including the thermal neutron, epithermal neutron, and fast neutron while the TLD detector more focuses to count the slow neutron (thermal neutron and epithermal). Both TLD-600 and TLD-700 have more sensitivity towards thermal neutron with more than 86% [17]. Besides that, according to Devine [18], the Microspec- 6 N-probe give a coarser spectrum due to the significant bias in term of energy between the transition region of fast and thermal that caused the spectrum in many cases. The measurement by using both Microspec-6 and TLD demonstrated the same pattern.

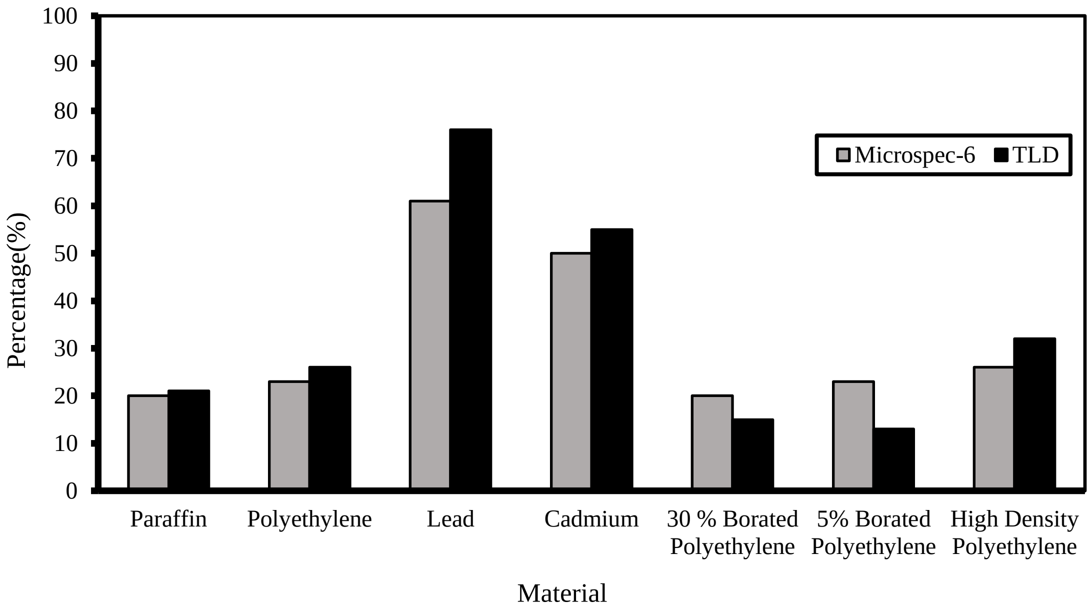

Figure 8. The percentage of gamma produce by using Microspec-6 and TLD detector.

According to result and discussion from both experiment, paraffin and 30% borated polyethylene were chosen for further research based on their characters as the neutron moderator and neutron collimator. Even though 5% borated polyethylene have same characteristic with 30% borated polyethylene, but due to the less of boron-containing on the polyethylene have make 5% borated polyethylene was least choice for the BNCT collimator design. The presence of boron as the neutron absorber could controlled the nuclear reactions [19]. In terms of material shielding, lead and HDPE was chosen for further research. The lead was chosen due to the characteristic of gamma shielding and have high thermal neutron peak that can be used to increase the thermal neutron intensity for BNCT purpose. On other hands, HDPE was selected based on the performance of reducing fast neutron and high neutron attenuation coefficient. Besides that, HDPE also have strong moderation characteristic also in contrast, the cadmium was least choice due to the higher cost of cadmium itself and the cadmium have lowest thermal neutron peak which is suitable for produce high epithermal neutron for BNCT purpose. The collimator design on this research for BNCT purpose was aim to focus on a beam of thermal neutron compare to the epithermal neutron.

3. Design of BNCT collimator using MCNP simulation

Subsequently the selection of material based on the material characterization toward the neutron being finalized. All the material selected are used in the designing of collimator for BNCT purpose based on their characterization towards neutron and gamma. From all the material used in designing collimator, only aluminium is fixed as a casing for the collimator because of the low activation energy characteristic compare to other material. According to Chandler [20], aluminium is commonly and widely used in nuclear reactor and nuclear applications such as nuclear waste storage and etc. There are 10 designs of collimator geometry are shown as figure 9 by using VisEd software and table 7 for summarization of material used for collimator designed.

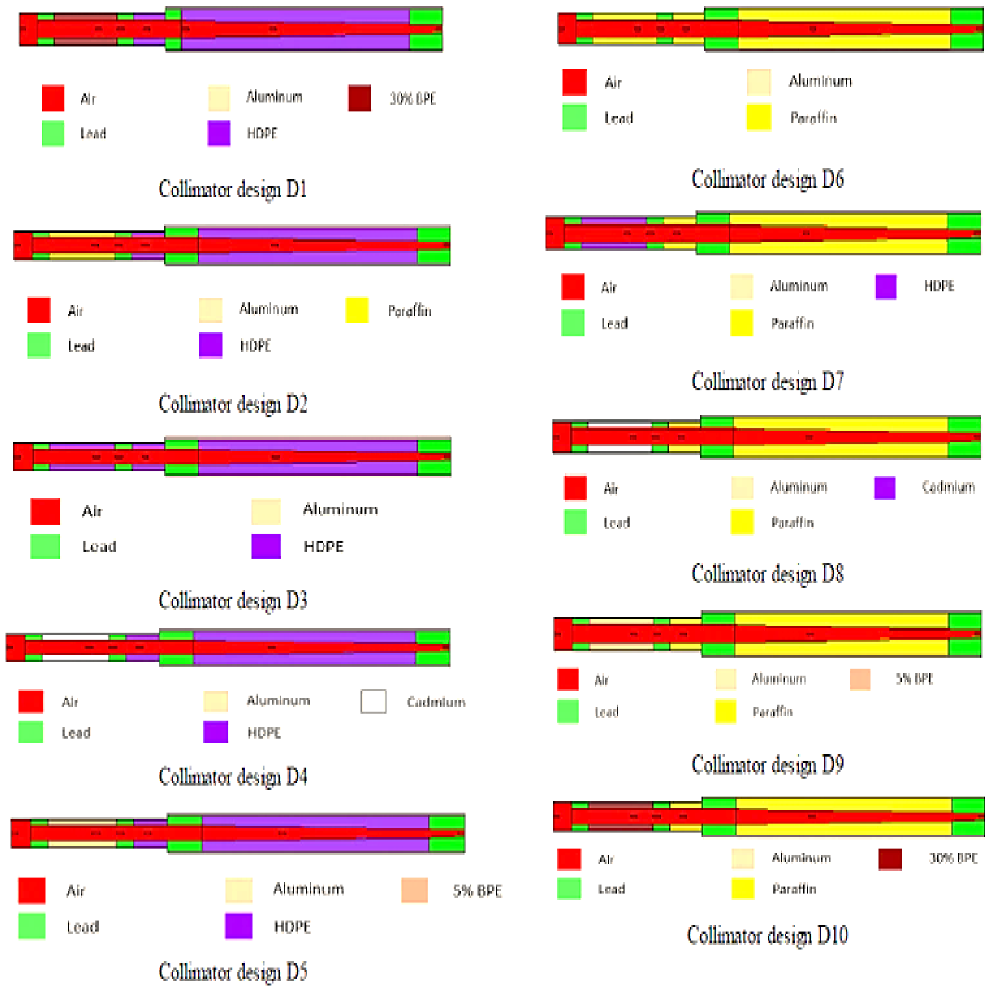

Figure 9. The collimator design D1-D10.

Table 7. Summarization of materials used in collimator.

No. of Collimator Design	Material
D1	HDPE + Aluminium + Lead + 30% BPE
D2	HDPE + Aluminium + Lead + Paraffin
D3	HDPE + Aluminium + Lead
D4	HDPE + Aluminium + Lead + Cadmium
D5	HDPE + Aluminium + Lead + 5% BPE
D6	Paraffin + Aluminium + Lead
D7	Paraffin + Aluminium + Lead + HDPE
D8	Paraffin + Aluminium + Lead + Cadmium
D9	Paraffin + Aluminium + Lead + 5% BPE
D10	Paraffin + Aluminium + Lead + 30% BPE

IOP Conf. Series: Materials Science and Engineering **736** (2020) 062023 doi:10.1088/1757-899X/736/6/062023

The shielding materials used in the collimator were selected based on the characterization using Microspec-6 and TLD. It is divided into two types of shielding material which is neutron shielding materials and gamma shielding materials. Neutron shielding materials consist of paraffin and high-density polyethylene (HDPE) while gamma shielding materials consist of lead. For neutron collimate material, different material was used in this collimator design. Figure 10, show the cell card design of collimator using MCNP. The input source for this collimator design is cell number 2 and the output source was at the number 21 cell. The simulation of MCNP result for all the collimator design was tabulated at table 5 and figure 11 for neutron flux produce from the collimator.

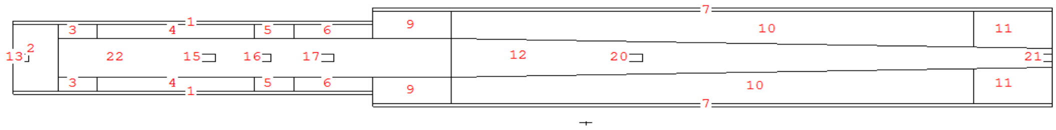

Figure 10. Collimator cell card.

Based on table 8 and figure 11, the simulation shows that most of the collimator design have the same range of neutron flux (thermal, epithermal and fast). For the thermal neutron flux, average flux produce was obtained was 3.75 x 10^9 neutron.cm^{-2}s^{-1} and the highest thermal neutron produced was from design D1 that comprise of lead, 30% borated polyethylene, HDPE and aluminium. All the design used HDPE as collimate material (D1, D2, D3, D4 and D5) simulate with higher thermal neutron compared to the paraffin-based design (D6, D7, D8, D9 and D10) with 7.8% of relative different between the averages. In addition, the design of collimator using HDPE also obtained higher epithermal neutron flux compared to paraffin with only 7% of different percentage. For epithermal neutron flux, the used of HDPE on D3 had resulted 1.77 x 10^7 neutron.cm^{-2}s^{-1}. The strong moderation characteristic from paraffin lead to most of the design used paraffin as neutron collimate material was obtained lower epithermal flux compared to the HDPE. Besides that, the effectiveness of HDPE as neutron fast shielding and moderator was proved on the fast neutron flux measurement. All the design from D1-D5 simulate low fast neutron flux compared to paraffin (D6-D10). In conjunction, the design D1 was qualified for BNCT collimator based on the MCNPX resulted on highest thermal neutron flux and epithermal neutron flux compared to other design even though there are less than 1% different of fast neutron with the highest design.

Table 8. Simulation results of neutron flux across the beam line of BNCT collimator using different material.

Collimator design	Neutron Flux, neutron.cm^{-2}s^{-1}		
	Thermal	Epithermal	Fast
D1	1.5770 x 10^9	1.7744 x 10^7	9.7484 x 10^6
D2	1.5599 x 10^9	1.7744 x 10^7	9.7476 x 10^6
D3	1.5611 x 10^9	1.7751 x 10^7	9.7486 x 10^6
D4	1.5610 x 10^9	1.7750 x 10^7	9.7317 x 10^6
D5	1.5610 x 10^9	1.7750 x 10^7	9.7313 x 10^6
D6	1.4997 x 10^9	1.6994 x 10^7	9.6319 x 10^6
D7	1.4998 x 10^9	1.6994 x 10^7	9.6325 x 10^6
D8	1.4997 x 10^9	1.6994 x 10^7	9.6326 x 10^6
D9	1.4999 x 10^9	1.6994 x 10^7	9.6327 x 10^6
D10	1.4997 x 10^9	1.6994 x 10^7	9.6345 x 10^6

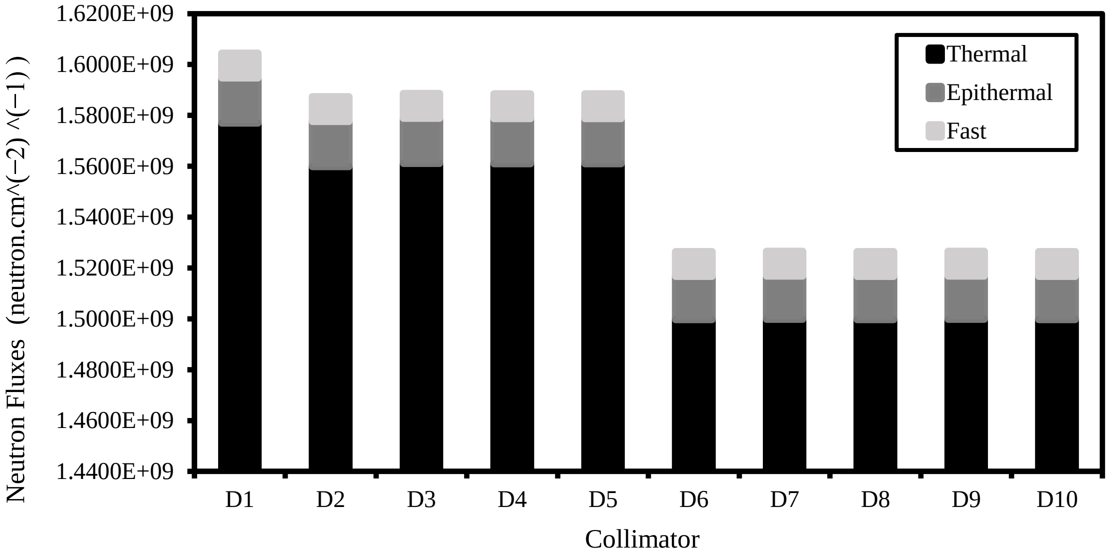

Figure 11. Graph of neutron flux across the beam line of collimator with different material.

4. Conclusion

After determining the material characteristic towards neutron at the thermal column of RTP, research was continued to identify the suitable material for BNCT collimator design. There are seven materials being tested which are paraffin, polyethylene, cadmium, lead, 30% borated polyethylene, 5% borated polyethylene and HDPE (high density polyethylene). All the samples have the same thickness of 5 cm and was irradiated by RTP and dose reading were measured TLD and Microspec-6 detector. The characteristic of material was observe based on their behaviour towards thermal neutron peak, attenuation coefficient and neutron spectrum profile. Lead provide highest thermal neutron peak with 5.81 x10^3n/cm2 and shielded more than 60% of gamma from neutron beam by both TLD and Microspec-6 detector. This research also found that HDPE was efficient to shield the fast neutron by more than 70%. Based on the measurement, the data was used to classify the samples material into shielding, collimate and neutron moderator for BNCT collimator design. Lead was expected to provide good shielding of gamma and HDPE has better shielding for fast neutron. In terms of neutron moderator, paraffin has sufficient moderation and collimation of neutron then followed by 30% BPE. The used of the ideal collimator based on D1 design have optimized the beam by increase the thermal neutron and the epithermal neutron at the end of collimator.

Acknowledgement

Authors would like to extend their gratitude to Ministry of Higher Education Malaysia and Universiti Malaysia Pahang (UMP) with grant number RDU190379.

References

[1] Solleh M R 2016 *Moderator, Collimator and Shielding Studies for BNCT Research atMalaysian Nuclear Agency* (Universiti Sains Malaysia.: Phd Thesis).
[2] Fauziah N, Widiharto A and Yohannes S 2015 *Tri Dasa Mega-Jurnal Teknologi Reaktor* **15(2)**
[3] Hawk, Andrew E, Blue, TE, & Woollard, JE 2004 *Applied radiation and isotopes* **61(5)** 1027-1031
[4] Shaaban, Ismail, & Albarhoum, Mohamad 2015 *Progress in Nuclear Energy* **78** 297-302
[5] Monshizadeh, M, Kasesaz, Y, Khalafi, H, & Hamidi, S 2015 *Progress in Nuclear Energy* **83** 427-432
[6] Solleh M. R. M., Tajudin, A. A., Mohamed, A. A., Munem, E. M. E.M. H., Yoshiaki, K 2011 *J. Nuclear Related Tech.* **8(2)** 41-48

IOP Conf. Series: Materials Science and Engineering **736** (2020) 062023 doi:10.1088/1757-899X/736/6/062023

[7]Durisi, E, Zanini, A, Manfredotti, C, Palamara, F, Sarotto, M, Visca, L, & Nastasi, U 2007 *Nuclear Instruments and Methods in Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipment* **574(2)** 363-369
[8]Arif MA 2016 *Study of Moderator (Collimator) Materials for Boron Neutron Capture Therapy(BNCT) Facilities at Thermal Column RTP* (Uniten : Degree Thesis)
[9] Faghihi, F, & Khalili, S 2013 *Radiation Physics and Chemistry* **89** 1-13
[10] Akan Z, Türkmen M, Çakır T, Reyhancan İ A, Çolak Ü, Okka M and Kızıltaş S 2015 *Applied Radiation and Isotopes* **99** 110-116
[11] Solleh M R M 2015 Moderator, collimator and shielding studies for BNCT research at MalaysianNuclear Agency (Universiti Sains Malaysia: PhD Thesis)
[12] Zafiropoulos D, Scarabottolo G 2001 *Neutron Spectra using a portable Neutron Spectrometer*
[13] Bubble Technology Industries Inc 2014 Spectroscopic Neutron Probe. Available from:http://bubbletech.ca/wp-content/uploads/2014/02/BTI_MICRO_Neutron-Probe_2011-Feb-26.pdf (2 August 2018)
[14] International Atomic Energy Agency 2001 Current status of neutron capture therapy (IAEA:TECDOC-1223)
[15] Obaid S S, Sayyed M I, Gaikwad D K and Pawar P P 2018 *Radiat. Phys. Chem.–* **148** 86–94
[16] Da Silva A., Crispim V 2001 *Appl. Radiat. Isot.* **54(2)** 217–25
[17] Paiva F, Siqueira P T D and Cavalieri T A 2015 Comparing the responses of TLD 100, TLD 600,TLD 700 and TLD 400 in mixed neutron-gamma fields *INAC 2015: International NuclearAtlantic Conference Brazilian Nuclear Program State Policy For A Sustainable World*
[18] Devine R T, Romero L L, Gray D W, Seagraves D T, Olsher R H and Johnson J P 2002 *Nucl. Instruments. Methods Phys. Res. A. Accel. Spectrometers, Detect. Assoc. Equip.* **476(1–2)** 416–22
[19] Korkut T, Karabulut A, Budak G, Aygün B, Gencel O, Hançerlioğulları A 2012 *Appl. Radiat. Isot.* **70(1)** 341–5
[20] Chandler D L 2016 *Aluminium used in nuclear reactors and other harsh environments may lastlonger with new treatment*