STUDY OF MODERATOR (COLLIMATOR) MATERIALS FOR BORON NEUTRON CAPTURE THERAPY (BNCT) FACILITIES USING SANS

Anas Muhamad Pauzi[1], Muhammed Ariff Mohd Sarif[1], Abdul Aziz Mohamed[1], Faridah Idris[2]

[1]*College of Engineering, Universiti Tenaga Nasional, Jalan IKRAM-UNITEN, 43000 Kajang, Selangor, Malaysia.*

[2]*Malaysian Nuclear Agency, Kompleks PUSPATI, 43000 Kajang, Selangor, Malaysia.*

Corresponding author: anas@uniten.edu.my

ABSTRACT

The Malaysian TRIGA Mark II reactor was investigated for this research. The objective of this research is to study the moderator (collimator) materials for Boron Neutron Capture Therapy (BNCT) facilities proposed. The experiment was conducted by using Small Angle Neutron Scattering (SANS) beam as the neutron source. The materials use as sample was Steel slab, Aluminum slab, Cadmium slab, Lead slab, Boron Carbide (B_4C), Plasticine Clay, Water, Concrete slab, Polyethylene and Paraffin. The samples were irradiated by using SANS neutron and the fluxes of neutron with energies were measured using MICROSPEC-Spectrometer detector. The neutron spectrum profiles for each sample were recorded and compared for the purpose of analyzing the materials characteristic. Based on the results, Paraffin possesses the highest thermal peaks, followed by Concrete and Steel. The experiment result analysis was continued with neutron penetration and attenuation coefficient measurement for each samples. Cadmium slab shows the highest attenuation coefficient value, follows by Polyethylene and Water respectively. Neutron cross section analysis was also conducted based on the data collected from the experiment. The highest scattering cross section is demonstrated by Water followed by Polyethylene. The lowest was Paraffin. The highest absorption cross section is demonstrated by Cadmium and the lowest was Boron Carbide. Weight ratio used to determine the material characteristics on acting as the good neutron moderator. Polyethylene possesses the highest ratio, followed by Water and Paraffin. Based on results and analysis, BNCT facility is proposed to be consisting of three different materials, combining of Paraffin, Polyethylene and Water. Neutron shielding elements surrounding the collimator should be a combination of Cadmium and Boron Carbide. The structural frames of BNCT facility is suggested to be consist of Concrete, Steel and Aluminum, due to the materials capabilities and the cost factor.

ABSTRAK

Reaktor TRIGA Mark II Malaysia telah disiasat untuk penyelidikan ini. Objektif penyelidikan ini adalah untuk mengkaji bahan moderator (kolimator) untuk kemudahan Boron Neutron apture Therapy (BNCT) yang dicadangkan. Eksperimen ini dijalankan dengan menggunakan rasuk Kecil Angle

Neutron (SANS) sebagai sumber neutron. Bahan-bahan yang digunakan sebagai sampel adalah Slab Steel, Slab Aluminium, Slab Cadmium, Papak Lead, Boron Carbide (B4C), Plasticine Clay, Air, Papak Konkrit, Polietilena dan Parafin. Sampel tersebut disinari dengan menggunakan neutron SANS dan fluks neutron dengan tenaga diukur dengan menggunakan pengesan MICROSPEC-Spectrometer. Profil spektrum neutron bagi setiap sampel telah direkodkan dan dibandingkan dengan tujuan menganalisis ciri-ciri bahan. Berdasarkan hasilnya, parafin mempunyai puncak termal tertinggi, diikuti oleh Beton dan Baja. Analisis hasil eksperimen diteruskan dengan penembusan neutron dan pengurangan pekali pengurangan untuk setiap sampel. Slab kadmium menunjukkan nilai pekali pelemahan tertinggi, mengikut Polyethylene dan Air masing-masing. Analisis keratan rentas Neutron juga dijalankan berdasarkan data yang dikumpulkan dari eksperimen. Bahagian keratan rentas yang paling tinggi ditunjukkan oleh Air yang diikuti oleh Polyethylene. Yang paling rendah ialah Paraffin. Bahagian penyerapan tertinggi ditunjukkan oleh Kadmium dan yang paling rendah adalah Boron Carbide. Nisbah berat yang digunakan untuk menentukan ciri-ciri material yang bertindak sebagai moderator neutron yang baik. Polietilena mempunyai nisbah tertinggi, diikuti oleh Air dan Paraffin. Berdasarkan hasil dan analisis, kemudahan BNCT dicadangkan untuk terdiri dari tiga bahan yang berbeda, menggabungkan Parafin, Polietilena dan Air. Unsur-unsur pelindung Neutron yang mengelilingi kolimator itu mestilah gabungan Kadmium dan Boron Carbide. Bingkai struktur kemudahan BNCT disarankan untuk terdiri daripada Konkrit, Keluli dan Aluminium, kerana keupayaan bahan dan faktor kos.

Keywords: BNCT, TRIGA, neutron, moderator

INTRODUCTION

The study is to contribute a data source for development of BNCT facility at thermal column of Reactor TRIGA Puspati Malaysia. The main objectives were to investigate and understand the materials specification for the BNCT neutron collimator at Thermal Column and also the neutron shielding characteristics of the chosen materials. Hence, suitable materials that can be used to fabricate the neutron collimator, neutron shielding and the structural frame of BNCT facility. The study will be narrowed to explore the ability, quality and availability of materials to be utilize as neutron collimator, neutron shielding and structural frame for the purpose of BNCT facility

Theory of Boron Neutron Capture Therapy (BNCT)

Boron Neutron Capture Therapy, BNCT is widely developing in medical industry as a promising treatment method for malignant brain tumor (Byung Chul Lee, 2005). The number of cases for this type of cancer is increasing fast and it courage the scientist to develop an alternative for treating this malignant tumor. By improving the result and efficiency of the ordinary standard treatment of brain tumor, BNCT offer highly selective destruction of tumor cells with minimal damage to the healthy cells surrounding the tumor area (Munem, 2007).

The mechanisms of this precise therapy are simple and safe. Boron (^{10}B) carrier compound is usually deposited firstly to patient by intravenous infusion. In the absence of neutron, the boron compound is a non-toxic and non-radioactive agent that accumulates into cancer cells. After a sufficient time, approximately 2 to 2.5 hours, boron compound is optimally absorbed in the tumor cells while the healthy tissue has lesser boron concentration. The tumor area is irradiated with neutron at a sufficient energy, which thermalize in tissue and interact with the ^{10}B nuclei. High linear energy transfer (LET) resulting of alpha and lithium particles are produced, hence damaging the cancer cells (Savolainen, May 2012).

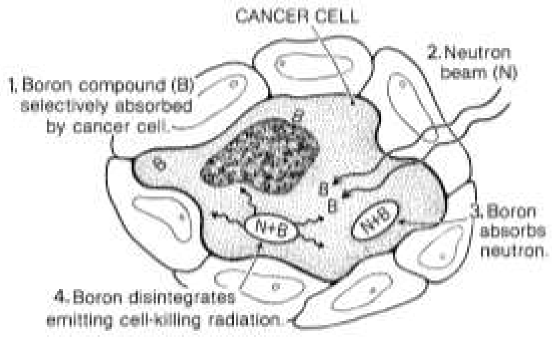

Figure 1. BNCT treatment cycle

Theoretically, it is possible to selectively irradiate the tumor cells that has absorb a sufficient amount of 10B compound and simultaneously spare the healthy cells. The basic nuclear reaction of the therapy is given by

$$
{}^{10}_{5}B + {}^{1}_{0}n \rightarrow {}^{11}_{5}B* - \begin{cases} \boxed{{}^{4}_{2}He + {}^{7}_{3}Li + 2.79\ MeV\ (6.1\%)} \\ \boxed{{}^{4}_{2}He + {}^{7}_{3}Li + \mathbf{2.31\ MeV}\ (93.9\%)} \\ \quad \boxed{{}^{7}_{3}Li + \gamma\ (0.48\ MeV)} \end{cases} \tag{1}
$$

The energy of the neutron used is depends on the depth and penetration of the tumor (Florberg, 2002). For thermal neutron, less than 0.005eV neutron flux were used to irradiate superficial tumors, and epithermal neutrons, from 1eV to 10,000eV neutron flux were used for deep seated tumors due to the penetration is deeper into the body before being slowed down to thermal energies. This process allowed a very high, localized dose of ionizing radiation that is efficiently destroy the cancerous cells compared to traditional ionizing radiation, assuming boron is selectively concentrated in the cells.

Neutron Analysis

The cross section for each material is evaluated by author in this project. The concept of cross section is defined as the probability of a particular event occurring between neutron and a nucleus. Cross section has the dimension of area, called barn. Macroscopic cross section would be highlighted in this research due to the mixture of element of each sample. The total cross section σ_t is consisting of scattering and absorption cross section for the elements. The updated value for both cross sections is referred through the Evaluated Nuclear Data File (ENDF), Brookhaven, England. There would be a slight difference on the value between other countries data such as Korean KAERI and France OECD Nuclear Energy Agency (NEA). It is valid to use either one of them and still can be considered as relatively the same. The elastic scattering macroscopic cross section from each sample is calculated and lead to understanding the interactions of a neutron passes through a thick layer of matter and the moderation of neutron speed. In elastic scattering, the total kinetic energy of the neutron and nucleus is unchanged by the interaction. During the interaction, a fraction of the neutron's kinetic energy is transferred to the nucleus.

For this project, one of the important theories that need to understand is the build-up factor. Build-up factor are geometry dependent. The fraction of the scattered component would be different if the shielding material changes the shape. It would also be different if the source beam, instead of being perpendicular, would approach the shield at some angle. The build-up factor are also dependent on the energy of the source radiation. The correction for the attenuation of the un-collided beam requires the use of a linear attenuation coefficient, which in turn depends upon the source beam energy. Need to be understood that various type of building factors can be defined. There is not a single set of build up factors but a considerable range of them.

METHODOLOGY

Early on the planning stage, the proposed material to be use as sample for this experiment was Bismuth-209, Polyethylene and Lead. The material selected has been studied to ensure that it is relatable and synchronized with the project objectives. Due to the unexpected amendment for this experiment, the material selected was changed to ensure the availability and mobility of this project. The material used for this experiment will be discussed in the next subtopic. Neutron released from SANS neutron collimator will be use as the neutron source for this experiment. It would substitute for the thermal column, as it is unavailable for this period due to the reactor is in maintenance schedule and time constraint the author face to prepare the experiment. Materials list and specification used in this study are shown in Table 1.

Table 1. Materials list and specification

Materials	**Dimensions, H x W (cm)**	**Thickness (cm)**
Steel (Mild) Slab	20 x 15	0.5
Aluminum Slab	21 x 15	0.5
Cadmium Slab	25 x 17	0.5
Lead Slab	20 x 15	0.5
Polyethylene Slab	27 x 16	0.5
Boronated Carbide Sheet	10 x 10	0.5
Paraffin Slab	8 x 6	0.5
Concrete Slab	21 x 21	0.5
Plasticine Clay	4 x 4	0.5
Water	10 cm (diameter)	-

The experiment were carried out as follows:

i. Preparation for experiment apparatus and samples. Wear proper attire and dosimeter. Sample holder and MICROSPEC-2 Spectrometer were prepared at the table provided.
ii. Check for any irregularities at experiment area, make sure SANS beam were cleared and ready to be use.
iii. Switch on the MICROSPEC-2 Spectrometer for warm-up purpose. Approximately 30 minutes required.
iv. Reactor start-up and command from reactor team. Operate at maximum 750kW.
v. Prepare sample at sample holder. Detector and the analyzer is in ready mode.
vi. Open the beam port valve slowly until fully open.
vii. The samples were then irradiated and time were recorded. In this case, the irradiation time were between 5 to 6 min.
viii. Save the data in the analyzer.
ix. Slowly close the beam port valve until fully closed.

x. Repeat procedure 5 to 8 for the rest of the samples.
xi. Data saved in the analyzer were double checked.
xii. Finish the experiment, inform reactor team and slowly closed the beam port valve.
xiii. Reactor shut down. Clear up the surrounding area and housekeeping.

The data saved in the analyzer must be unfold by using N-View software. Then, the unfolded data can be transform into spectrum profile by using Veusz software. The experimental setup will be categorized accordingly in Table 2.

Table 2. Description of experiment component

Components	Functions	Set-up	Material
Monochromator	Reflects neutron from reactor for an average wavelength (0.5nm)	As provided	Pyrolytic Graphite
Neutron collimator	To collimate the incoming neutron to illuminate the samples	4m long x 100mm diameter, as provided	Shielded Steel Tube
Sample	Experimental purposes	Samples holder; steel block as provided	*refer table 1
MICROSPEC 2 Detector	Measure the intensity of neutron doses in the mixed field environment; through samples	As Provided	Liquid scintillator, helium^3 detector
Sample holder table	Sample and detector position; after collimator	As provided	Stainless Steel

Figure 2, 3, 4 and 5 show the schematic and actual experimental set-up of the study.

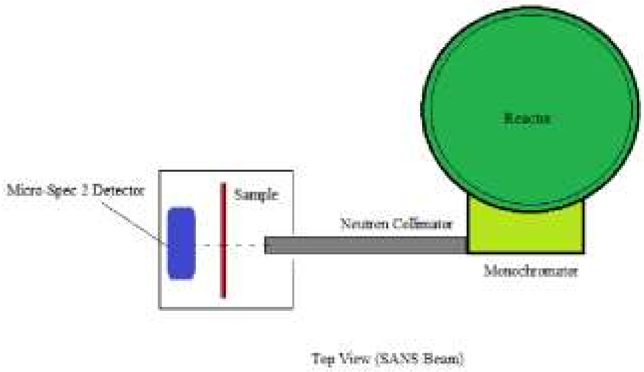

Figure 2. Experimental setup illustrated (Top View)

Figure 3. Experimental Setup (Actual Picture)

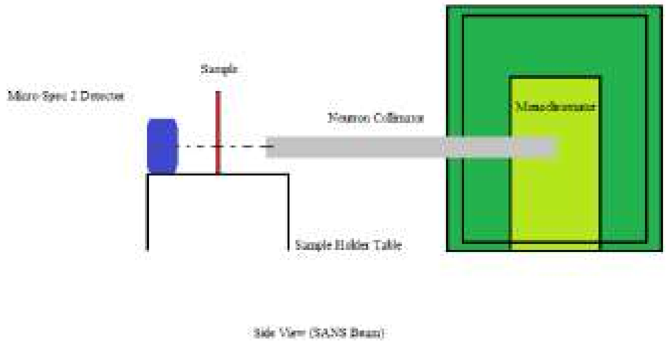

Figure 4. Experimental setup illustrated (Side View)

Figure 5. Experimental Set up (Beam view)

RESULTS AND DISCUSSION

Results were obtained to analyze the energy spectrum at repeating open source, then analyzing effects of moderating materials and gamma shielding materials and finally neutrons penetration were calculated.

Neutron Spectrum Profiles

Figure 6 shows the variation in results of neutron fluence on an open source, at three different reactor run on the same power. The average of the fluence were calculated and plot as below. It shows that the neutron spectrum detected shows high thermal neutron flux of between 10^4 to 10^5, and the flux ranges all the way to fast neutron spectrum. Although the significant amount of uncertainty obtained, the results still shows a consistent trend that will be useful for further analysis.

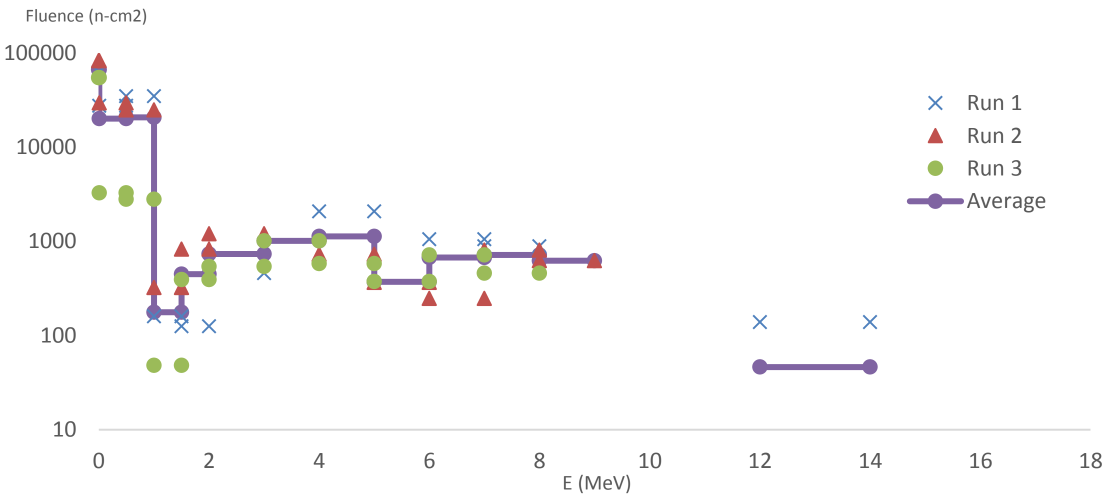

Figure 6. Neutron fluence for open source at different runs

Figure 7 shows the experimental results of neutron spectrum for neutron moderating materials which are water, paraffin and polytheline. All three materials had moderate the fast neutron from 10^{14} MeV to at least 10^{10} MeV, where Polytheline and Paraffin shows stronger moderating effect compared to water, which is a maximum energy of 10^8 MeV. Water had shown to also reduce the intensity of neutron fluence at the thermal neutron to about 10^4 n-cm^2, while paraffin increases the neutron fluence for thermal neutron to 10^5 n-cm^2. Overall, this shows the importance of paraffin in reducing fast and epithermal neutron while maintaining the intensity of thermal neutron, which suits the moderating function of the BNCT collimator.

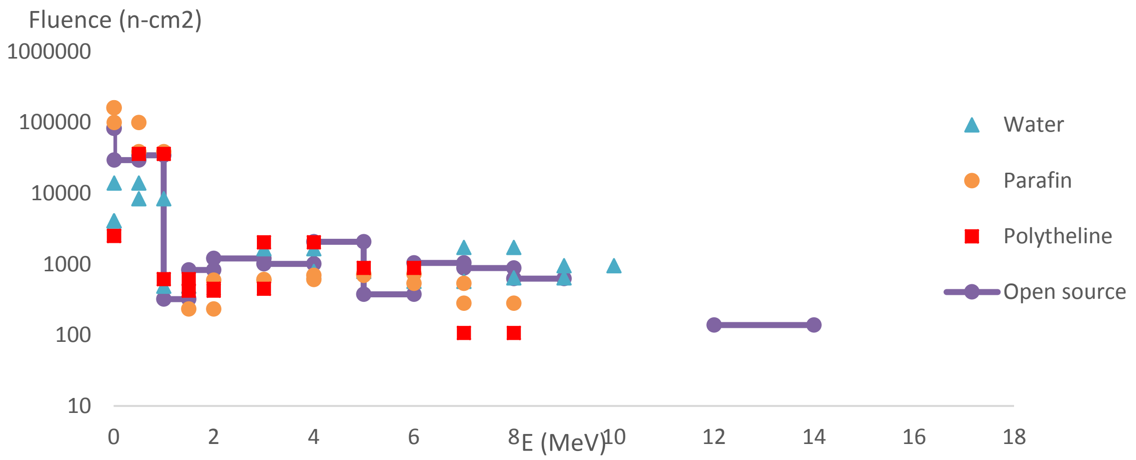

Figure 7. Neutron flux spectrum of moderating materials

Figure 8 shows the results of neutron flux for gamma shielding materials which are Aluminium, Steel, Cadmium, and Plastecine. Observing at above 10^{12} MeV shows that only Cadmium would not cause any moderation on the neutron, but overall it reduces slightly the neutron intensity. Steel is also good material for BNCT moderator because it maintains the intensity of the neutron at thermal neutron region.

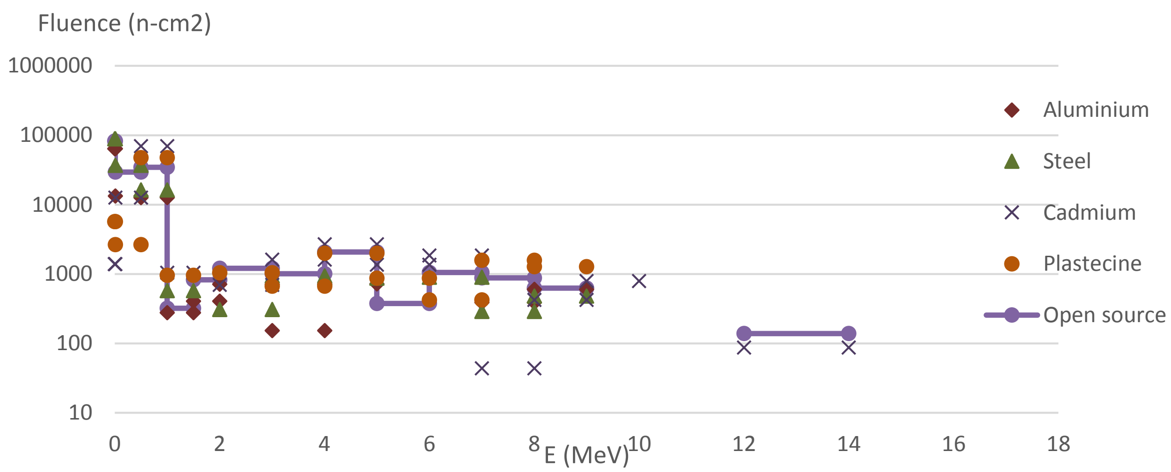

Figure 8. Neutron flux spectrum for gamma shielding materials

Neutron Penetration

The decreasing of neutron intensity for each sample was recorded in this experiment. By using the recorded value, author can calculate the neutron attenuation coefficient for each sample. This will lead author to understand the characteristic of the material and selecting the effective shielding material. Usually, the decrease of neutron intensity by shielding materials is proportionally equal to the decrease in gamma ray intensity (Raffi, 2011).

For calculating the neutron attenuation coefficient, it is given by the equation;

$$\boldsymbol{I_d = I_o\, e^{-\mu x}} \tag{2}$$

where I_d is the neutron intensity after passing the materials (recorded value), I_o is the initial neutron intensity. The unit for both of the value is in n/cm^2. Neutron attenuation coefficient, μ is in m^{-1} (per meter) and d, the material thickness is in meter.

The value for I_o for each sample is 8.2x10^4 n/cm^{-2}, by using the value recorded for open beam measurement. The thickness for each sample is 0.5cm (0.005m), standardize for all material.

Table 3. Neutron intensity of each material with neutron attenuation coefficient

Materials	Neutron intensity Id (n/cm2)	Attenuation coefficient µ(m-1)
Cadmium	1.3x10^3	828.9
Polyethylene	2.4x10^3	706.3
Water	4.0x10^3	604.1
Borated Carbide (B_4C)	4.4x10^3	585.1
Plasticine Clay	5.6x10^3	536.9
Aluminum	6.3x10^4	52.8
Lead	8.2x10^4	0
Steel	8.8x10^4	-14.2
Concrete	1.2x10^5	-76.2
Paraffin	1.6x10^5	-133.7

Based on Table 3, the neutron attenuation coefficient for cadmium is the highest among other material. Cadmium attenuates most of the thermal neutron from the beam. It follows by polyethylene and water. The negative sign shows that the material does not attenuate the neutron but increasing the neutron fluences. It is might be a sign of an error, but the value shown is proportional with the neutron spectrum measured for the materials.

Source of Error

The variation in spectrum profiles for each material had been analyze for continuation of the project. Each sample shows variation in neutron intensity, as the energy level is increasing, the intensity is decreasing. This is a sign that every material tested in this experiment is relevant and suitable for the purpose of the objective, which is to study and to propose the most suitable material to be utilized as collimator.

Value for neutron attenuation coefficient for each sample is calculated. This will help to analyze and propose a good shielding material at BNCT facility. BNCT Facility is proposed to be constructed at reactor thermal column and shielding issue must be highlighted in the future research to ensure we build a safe and efficient BNCT facility for the patient.

For the selection of collimator material, based on macroscopic cross section calculation for each sample that has been proved in the table above, selection of suitable collimator material has been narrow down. It is hard to just rely on single material for the collimator. Several issue need to be considered such as scattering, absorption, secondary radiation produced, material availability and cost.

For structural material, the design would be large and complex compared to the collimator. Material with the proved shielding efficiency is required, as well as relevant in cost. Future research must be conducted to investigate more deeply on this issue and provide data for future development of BNCT facility.

The Microscopic Spectrometer Detector. The detector is consisting of 2 detectors working in unison, as mention previously. As it can be seen from the spectrum, there is an area where there is no reading can be detected. It is because of the dead time of the detector. The dead time is the time after each event during which the detector is unable to record another radiation (neutron pulse). The agency only have one set of this type of detector and according to the CO-SV and researcher, the efficiency of the detector is decreased as it has not been calibrated for quite amount of time.

The material conditions and calculation error. Author was unable to prepare the sample perfectly, such as the dimension and thickness. It was due to time constraint, effect of project dynamic (switch from thermal column to SANS beam) and reactor maintenance schedule. Apart from that, the calculation might be having some error as the value use from the nuclear data might be not accurate for the material composition that has been tested. This issue must be overtakes in the future research of BNCT facility development

Lastly is the SANS beam efficiency. As the reactor is in maintenance schedule, it is possible that the neutron source coming from SANS beam is not stable and inefficient as it should do. It might be too small for a possibilities, but it is the least point of the author critical thinking can go for investigating the source of error for this experiment.

CONCLUSION AND FUTURE RECOMMENDATION

The experiment conducted in this research was a success and reach the target. It was intensely done with small number of material and available equipment. Even though it has undergone minor dynamic at the early stage, author manage to adapt with it and improvise the research. Minimal time allocated for author to use the reactor especially due to the maintenance work schedule and weather condition. For the future, this research can produce a result with larger scope if the specified issue can be addressed.

Polyethylene is the most suitable material for neutron collimator in BNCT facility. It is also found that Polyethylene is the best neutron moderator, compared to the other samples. Extensive research with holistic methodology is suggested to be done to justify this finding. Apart from that, Water and Paraffin is also an option to be consider as a neutron collimator and can be further investigated.

Cadmium is suitable neutron shielding material in this experiment. It can be used to isolate the collimator, by a thick layer, as to form a shielding from loss of thermal neutron. The intensity of thermal neutron penetrated to patient body need to be regulated to ensure the effectiveness of the treatment. It also can be combined with lead to attenuate gamma exposed outside the BNCT facility. Boron Carbide is also proved suitable to be consumed as neutron shielding material, as well as Steel. More research needs to be conduct to ensure the reliability of the data from this experiment.

The structure for BNCT facility would be large and complex. It will also consume the largest portion of material, compared to the neutron collimator and neutron shielding materials. Based on that fact, Cadmium is the least choice even though it is the most effective neutron shielding material. Other option for structural material is Steel and Aluminum. It would help a lot in term of cost effectiveness but the major suitability and relevancy must be deeply investigate and should be further proposed for future development.

The MICROSPEC-2 Spectrometer Probe has provided a sufficient data for this experiment. Hence the result can be stated as encounter minor error and fault. This is proved by the large dead time value and dynamic graph trends for each sample. The issue must be highlighted and rectified to ensure the next user of this instrument would exclude from the defect. Also needed to be highlight is the position of the detector when measuring the neutron, the length between detector and materials, and also the beam aperture wide angle. These were also contributing to the data analyzed by the probe and lead to error in the results.

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