

A significant review on beam optimization for boron neutron capture therapy

S Shalbi¹, N Sazali^{1,2*}, and W N Wan Salleh³

¹Faculty of Mechanical Engineering, Universiti Malaysia Pahang, 26600 Pekan, Pahang, Malaysia.

²Centre of Excellence for Advanced Research in Fluid Flow (CARIFF), Universiti Malaysia Pahang, Lebuhraya Tun Razak, 26300 Gambang, Kuantan, Pahang, Malaysia.

³Advanced Membrane Technology Research Centre (AMTEC), School of Chemical and Energy, Faculty of Engineering, Universiti Teknologi Malaysia, 81310 Skudai, Johor Darul Takzim, Malaysia.

*azlian@ump.edu.my

Abstract. The thermal column at the TRIGA PUSPATI (RTP) research reactor can produce thermal neutron. However, the optimization on the thermal neutron flux produced should be performed to gain a sufficient thermal neutron for boron neutron capture therapy purpose. Thus, the objective of this review is to optimize the thermal neutron flux by designing the collimator with different materials at the thermal column. In order to fulfil the requirement, set by the IAEA standard, the study of BNCT around the world was being reviewed to study the suitable measurement, material, design, and modification for BNCT at the thermal column of TRIGA MARK-II, Malaysia. Initially, the BNCT mechanisms and history was review. Then, this paper review on the design and modifications for BNCT purpose around the world. Based on this review, suitable material and design can be used for the BNCT in Malaysia. Moreover, this paper also reviews the current status of BNCT at the RTP with the measurement of the thermal neutron flux was conducted along the thermal column at 250 kW. The thermal column of RTP was divided into 3 phases (Phase 1, Phase 2 and Phase 3) so that an accurate measurement can be obtained by using gold foil activation method. This value was used as a benchmark for the neutron flux produced from the thermal column. The collimator was designed using different types of materials, and their characteristic towards gamma and neutron flux was investigated. The reviewed demonstrated that the final thermal neutron flux produced was significantly for BNCT purpose. Lastly, this paper recommends the future research can be conducted on BNCT at RTP.

1. Introduction

Cancer is the most leading cause of death based on 172 countries in the world [1]. In fact, cancer had caused death over 8.2 million each year, which was roughly estimated about 13% around the world [2]. There are more than 100 types of cancer exist and each requiring different techniques of diagnosis. BNCT is one of the promising methods in order to cure cancer by using neutron source either from nuclear research reactor or neutron generator such as an accelerator. BNCT method used the combination of low energy of neutron (slow neutron) irradiation and the targeting of a tumour site injected with a proper boron containing compound. Basically, the tumour cell was not directly destructed by neutron, but indirectly destroyed by the results of nuclear reaction between neutron and boron. In theory, the BNCT also is one of the forms of radiotherapy which is selectively kills the cancer cells and less effect on other normal cell that used photon that will selectively deposited in tumour cell of boron carriers such as BPA and BSH. The standard boron carries compound must be enriched by Boron-10 about 20% natural abundance. The patient will be irradiated with slow energy of neutron to reach the ratio of high concentration of boron in the tumour cell [3]. The irradiation with slow neutron points out



of nuclear reaction that form the products of Boron-10 capture the thermal neutron and as a result of production Boron-10 convert into Boron-11 which decay from the emission of an alpha particle [4].

Initially in BNCT, thermal neutron is captured by Boron-10 and becomes unstable because of its properties towards neutron. Neutron, especially thermal neutron was well known for its neutral properties (no electrical charge) and could be absorbed by atomic nuclei which collide with it to create heavier isotope. The unstable reaction of Boron-10 and thermal neutron results of emitting both high LET (Linear Energy Transfer) α and γ before it changes to Lithium-7. The LET is a term that used to measure the force acting on a charged particle travelling through matter [5]. There are three factors affecting LET which is charge, velocity and the mass of the particle [6]. BNCT emitted both high LET α and γ which have deeper penetration in the range of $\sim 10\mu\text{m}$ which is approximately to the diameter of mammalian cell and promptly attenuating the radiation [7]. Hence, the BNCT give a highly localized treatment to the tumour cell compare to other treatment. The basic nuclear reaction is illustrated in Figure 1.

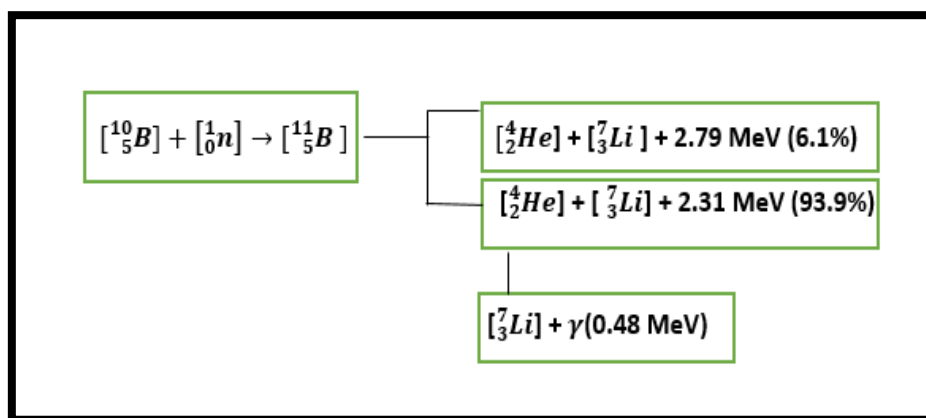


Figure 1. Neutron capture by the boron and subsequent nuclear reaction and nuclear fission [8].

The neutron produce from the research reactor was mostly use in the studies of BNCT around the world [8]. This kind of research are aim to develop a BNCT facility for the cancer treatment studies with safe and controlled from radiation and practically using slow neutron emitted from the Malaysia TRIGA MARK II research reactor (RTP). Thus, the first steps to achieve the long-term goal is to establish the suitability neutron fluxes produce from the research reactor. Commonly, most of the TRIGA reactor has a thermal column, which is the special place design to produce thermal neutron which can be utilized in BNCT facility. Therefore, the thermal column of RTP was identified to produce thermal neutron source in the investigation. In contemplation of developing the facility at the thermal column of RTP, the thermal neutron beam needs to be optimized by designing suitable ideal neutron collimator.

2. BNCT historical and review

After several years of disclosure of neutron by James Chadwick in 1932, Locher was firstly proposed the concept of neutron capture therapy. Locher discovered the availability of slow neutron capture with the high cross section naturally occurring isotope B-10 because of its properties that have high avidity in absorbing slow neutron. This is the initially described the BNCT in 1936 [9]. According to Chen [10], the uses of boron in BNCT was because of the properties of boron was highly selectively absorb by malignant tumour cells and it was mainly proposed of the destruction of cancer cells only. In theory, if boron could be selectively concentrated in a tumour and the volume then exposed to thermal neutrons, a higher radiation dose to the tumour relative to adjacent normal tissue would result [5]. This persuade the first clinical trials of BNCT in year 1950s-1960s using thermal neutron at Massachusetts General Hospital (MGH) and at the Brookhaven National Laboratory (BNL) and the trials were to identify the significant of the BNCT as an efficiency therapeutic treatment. The first clinical trial was considered as a failure because of the insufficient beam penetration but to curtail this problem, other trials was conducted at the Massachusetts Institute of Technology Research Reactor (MITR-I) with maximum

surgical of the tumour's precedent to BNCT. As the early stage of BNCT trials, there is no evidence of any survival using BNCT [11].

One of the challenges to enhance BNCT during those years is to develop and improve the new boron compound to achieve suitable tumour to brain concentration. Initially, in the early stage of BNCT studies, sulfhydryl–polyhedral borane (BSH) and borated phenylalanine (BPA) were used to improve the tumours to boron concentration ratio. This study was conducted by Dr Hatanaka, a neurosurgeon from Japan who being the apprentice to Dr Sweet at MGH (Massachusetts General Hospital) in 1968 after he returned to Japan. By using the BSH as a boron carrier, Dr Hatanaka had performed the open skull irradiation of the brain tumours using slow neutron [12]. After the success of the experiment in Japan, there are several groups of researchers working of BNCT using BSH for the brain tumour and as a result there are about 200 patients being treated with this technique with encouraging result [12]. At the same time, Dr Mishima a researcher from Japan also that working on borated phenylalanine (BPA) for treat malignant melanoma. Based on the documentation of Dr Mishima clinical trial and experiment on BNCT using BPA recorded that the normal skin of human is very response towards the BNCT treatment for malignant melanoma. In 1993, the beam use for BNCT had been more focus on intermediate neutron beam or epithermal neutron compared to the thermal neutron beam. The epithermal neutron was believed to have more penetration that can help in reducing the scalp reaction without having any complications. The MITR-II becomes the first who conducted the BNCT using the epithermal neutron to the human in 1994. The use of epithermal neutron for BNCT also had being widely use at the same time when the BMRR reactor apply to clinical trial for central nervous system tumours [11].

In Finland, the BNCT was conducted at the TRIGA MARK II type FiR1 reactor. The patients are intravenous administered with BPA-F as a boron carrier and was irradiated with the use of epithermal neutron without having other surgical procedures [13]. For the BNCT purpose, the neutron moderator material use is FLUENTAL which are the combination of 69% of AlF_3 , 30% of metallic Al and 1 % of LiF_7 . For the shield purpose bismuth was used to shield low gamma, thermal neutron and fast neutron contamination. The uses of the epithermal neutron beam for BNCT have been confirmed by the Finnish dosimetry team and verified by Idaho National Laboratory, Nuclear Research Institute (NRI) Republic of Czech and also Massachusetts Institute of Technology (MIT). According to [5], the advantage of using the TRIGA reactor for BNCT purpose is the stability and the availability to produce and supply the high neutron intensity with the low radiation background of the treatment beam. The clinical trials using TRIGA reactor had been done also at Europe, Japan and USA and none of the BNCT was postponed or cancel due to the reactor problem. The result of BNCT by using TRIGA reactor had convince the construction of BNCT for MARIA research reactor at the Institute of Atomic Energy at Swierk [14]. The schematic diagram of the FiR-1 BNCT facility has been show as Figure 2.

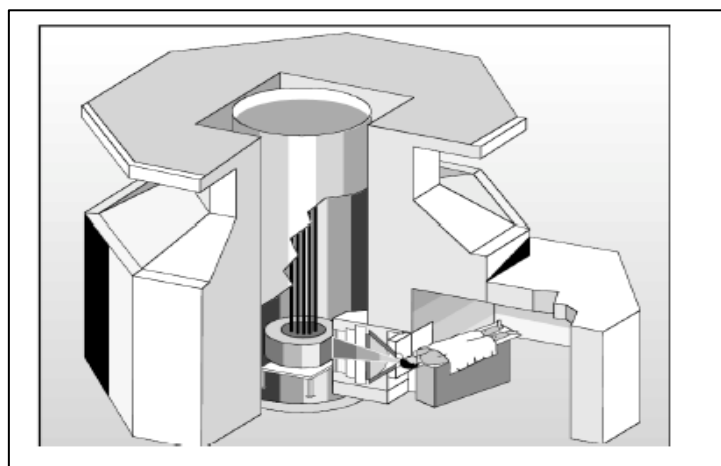


Figure 2. Illustration view of FiR-1 BNCT facility [15].

Despite of the research in BNCT by mostly government sector around the world, there are one private health company had looked forward of BNCT which is Boneca. According to Pohjola [16], the Boneca had proved their passion on BNCT by proved that the BNCT can be used to treat the head and neck cancer more beneficial compare to the conventional therapy. In fact, on 2007 there are clinical trials on 12 patients who suffering head and neck cancer with good response and result. About 7 patients had been successfully eliminated the tumour and the remaining patients had substantial tumour shrinkage with most of them have moderate adverse effect. To consider the good result, Boneca had looking forward by expand their study on BNCT for head and neck cancer for 30 more patients [16]. One of the reasons for choosing BNCT is the dose to the tumour can be increase without exceed the normal tissues tolerance [17].

In Turkey, the research of BNCT was conducted at the ITU (Istanbul Technical University) TRIGA MARK II research reactor. To produce high intensity and quality of neutron beam for BNCT purpose, the radial beam port has been modified. It has collimator from the material of polyethylene and Cerrobend installed. Cerrobend are the combination of 50% bismuth, 31.8% of lead and 18.2% of Sn. Figure 3 shows the geometry design of Polyethylene/ Cerrobend collimator.

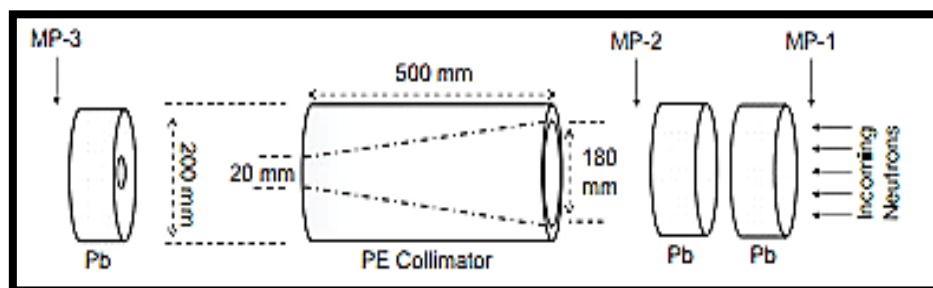


Figure 3. Geometric characteristic of Polyethylene/Cerrobend Collimator [18].

Before installing the collimator, the initial step must be used to measure the neutron flux produced by using neutron activation analysis (Au-Cd Foils). Then the results from the neutron activation analysis was used to verified using MCNP (Monte Carlo Neutron Particle) simulation. The used of polyethylene as collimator is good to reduce the slow neutron while the Cerrobend is more efficient to provide and produces the epithermal neutron [18]. Despite the use of research reactor as the neutron source for BNCT, the accelerator facility also is recognized as a neutron source. The accelerator is a machine that can be used to accelerate proton to the required energy for the reaction and results of neutron emitted at the energy range appropriate for BNCT [12]. The advantages of using accelerator are easy to operate, required less people to handle it compares to the research reactor, less overall risk than reactor and cheaper. The accelerator can be classified based on the range of low energy electrostatics machines to the higher energy cyclotrons. This accelerator however produces high energy neutron energy which is beyond the BNCT limit. Thus, the use of accelerator as a neutron source need the brute force moderation or large moderator assemblies for specifically obey the neutron specification for BNCT [19].

There are several studies around the world, including Argentina, Italy, Japan, Russia, United Kingdom and Israel regarding to development of AB-BNCT (accelerator base BNCT). The studies of the accelerator around the world consider on the different type of accelerator and the nuclear reaction. In Argentina for example, they investigated on the $^{13}\text{C}(d,n)^{14}\text{N}$ reaction as a neutron source for BNCT while in United Kingdom the reaction of $^7\text{Li}(p,n)^7\text{Be}$ are being studied as a neutron source to produce an epithermal neutron source for BNCT [20]. In terms of nuclear reaction, it can be divided into two different type which is endothermic reaction and exothermic reaction. The study on $^9\text{Be}(d,n)^{10}\text{B}$ reaction show that the ^9Be have advantages on thermo-mechanical properties ^7Li because it can lead the stabilization of residual products. Hence, become the advantages for hospitality-based facility because the $^9\text{Be}(d,n)^{10}\text{B}$ can be utilized about 1.4MeV, which implying the smallest possible accelerator compare to others [19]. Another neutron source can be used for BNCT purpose is a sealed

generator tube. The sealed generator tube is neutron generators which use the microwave ion source to produce the deuterium and tritium ions from the plasma. According to Solleh [5], the mechanisms of the sealed generator tube is by the microwave ions will yield a higher fraction of monoatomic ion species in the ion beam. Then the ions will accelerate to impinge the beam on the target with the energy of 14.1 MeV D-T neutrons that are generated from the fusion reaction. In fact, the yield of the neutron can be automatically controlled by the system and can have the stable yield also. Actually, the main objective of the development of a sealed generator tube is to study the feasibility of the neutron for BNCT to cure liver tumour. In procedures, there are four sealed generator tube use and produce about $4 \times 10^{11} \text{ ncm}^{-2}\text{s}^{-1}$ and two directly placed on each side of the patient's liver [19].

As mentioned above, the energy produced from the sealed generate or tube was 14.1 MeV is slightly high and cannot be used directly for destroying the tumour and the liver. Thus, the uses of neutron moderator were essential to produce the neutron energy in the intermediate form energy range. Different type of neutron moderator in the BSA (beam shaping assembly) are being provided which the neutron produce from the sealed generator tube must be passing through it to optimize the neutron for the BNCT purpose. Moreover, the direction of the neutron beam also needs to be guide and the study show that the reflector material such as lead can be used as a neutron reflector [19]. In terms of safety, the uses of the sealed generator tube as a neutron source is safer than the accelerator.

2.1. Neutron Properties

In 1932, James Chadwick discovered the subatomic particle called neutron while working at the Cavendish Laboratory. Neutron with symbol n or n^0 is identified with the characteristic of neutral (no electrical charge) and have a large mass comparable to proton. Neutrons are neutral and not affected by the electrons in an atom or by the positive charge of the nucleus. As consequences, neutron passes through the atomic electron cloud. In short, neutrons collide with nuclei but not with the atoms [21]. The kinetic energy of the neutron was significance in determining the nature of interaction between the neutron and the nucleus. Cacuci [21], stated that the interaction of neutron-nuclei will reduce the characteristic quantity of de Broglie wavelength $\lambda = h/2\pi$ of the neutron –nucleus centre of mass system which defined,

$$\lambda = \sqrt{\frac{h^2}{2ME}} \quad (2.1)$$

where), λ is the wavelength (m), M is the reduced mass (kg), E is the kinetic energy ($\text{kg m}^2 \text{s}^{-2}$) and h is the Planck constant.

2.1.1. Neutron absorption. Neutron absorption is defined as a neutron absorbed by the nucleus forming new particles. There are four type of neutron absorption which is radioactive capture, fission, charged particle emission and neutron emission. The neutron reaction cross section can be used to identify the probability of the neutron reaction with the consideration of one mono-energy neutron source interact with the nucleus which end up as a neutron flux, ϕ ($\text{neutron cm}^{-2} \text{s}^{-1}$). Equation 2.2 relate the relation of neutron flux, ϕ with n (neutron cm^{-3}) which is the number of neutrons per unit volume and its with neutron velocity, v (cm s^{-1}).

$$\phi = nv \quad (2.2)$$

$N \cdot dx$ (nuclei m^{-3}) is used when the target material contains N atom per unit volume have a certain width which has symbol as dx .

$$\sigma = \frac{c}{\phi N \cdot dx} \quad (2.3)$$

The equation above shows the neutron cross-section σ (barns), whereas C is belong to the neutron absorption (barns) [22].

2.1.2. Neutron shielding

In neutron shielding determination, the consideration of neutron intensity is very important to determine the best shielding material purposely to shield unwanted neutron. The intensity equation is given as:

$$I_d = I_o \cdot e^{-\mu d} \quad (2.4)$$

where I_d is the final intensity of neutron after the shielding (neutron $\text{cm}^{-2} \text{s}^{-1}$), I_o is the initial intensity (neutron $\text{cm}^{-2} \text{s}^{-1}$) of neutron before shielding, d is shielding thickness (cm) and μ is the neutron attenuation coefficient (cm^{-1}). In the deep penetration of fast neutron, equal of the fast neutron fluence is produced and the including fast neutron deep in a hydrogenous medium. Moreover, as the neutron energy decrease, the cross section of hydrogen in the high MeV energy region increases then lead to lower-energy neutron scatter than the high energy neutron. Basically, the attenuation of fission neutron is even practically by both hydrogen medium and non-hydrogen shield. The interaction of (n, γ) absorption cross-section of all of the medium used for shielding in the energy region was negligible and has no significant role in the removing the fast neutron because of the removed cross-section can be represent by the total fast neutron cross-section, an energy average of fission neutron and neutron scattering. Thus, to remove the cross-section is sufficient when the material used contain high hydrogen concentrations [5]. In addition, the fast neutron is attenuated very poorly with the non-hydrogenous material such as iron, lead and aluminium which is slightly used for photon shielding. Thus, the attenuation of fast neutron through this non-hydrogenous material can be neglected. The interaction between the fast neutron in the shielding design could produce the slow neutron and the intermediate neutron which also leads to the gamma production. In this research, one of the main aims is to successfully capture the gamma-ray dose at the surface of BNCT design with a proper consideration of shielding design and material. It is important to calculate and simulate the slow neutron and epithermal neutron in a shielding design for BNCT purpose.

One of the reasons for thermal neutron to be calculated is because of its properties of not being observed until the thermalization occurs. Moreover, during the process of the thermalization, the scattering can occur because of the addition of the fluences that become more isotropic. There is one approach that can be used in order to compute the build-up factor of low energy inside the shielding by using the combination of removal theory and multigroup diffusion theory. The combination of both theories, removal theory that defined as the penetration of fast neutron and multi-group diffusion theory which describe the subsequent thermalization and thermal diffusion lead to the contribution of the total dose at the shielding surface by the capture the gamma particle produced deep in the shielding and as a result of neutron absorption [5].

2.1.3. Neutron collimator

In this review, the most important part to be developed and studied is collimator. Collimator are well known as a beam limiting device that widely used in various industries including optical, medical and nuclear. Its acts as a filter of the beam allowing the beam in the specific direction to travel trough. In BNCT, the collimator inside the shielding purpose was to reflect the neutrons back into the beam [11]. In large neutron source such RTP, the neutron collimator important to reduce the neutron and gamma flux for specific purposes such as BNCT, neutron radiography, diffractometer and other application that use direct neutron source from the RTP. There are three common types of neutron collimator that widely used, namely are pinhole collimator, Soller collimator and divergent collimator. The neutron collimator must be properly design for the BNCT research facility. The thermal neutron flux emitted from the collimator used for BNCT must have good quality beam with the high intensity as required by IAEA standard. The use of collimator will avoid the over exposed of the thermal neutron, fast neutron, epithermal neutron and gamma ray to the patient. In order to design suitable collimator for BNCT research facility, the parameter of L/D ratio must be considered. The L/D is defined as the degree of divergence of the neutron beam where L stands for the length of the collimator while D is the diameter

of the entrance aperture [5]. To build the collimator, the combination of neutron collimator material, neutron shielding material and neutron moderator material specification are very important. In order to obtain these specifications, a simulation based on Monte Carlo N-Particle Transport Code (MCNP) method can be used to generate the suitable parameters for neutron collimator.

2.1.4 Neutron moderator

In order to optimize the neutron flux beam into the maximum output. Most of the fast neutron produce at the thermal column should be material that moderated using low atomic mass, does not produce moist, does not decompose in high radiation field and the neutron activation product produce should be short in half life. Commonly, in new world, the suitable material use to neutron moderator are Al, Al₂CO₃, Al₃F and D₂O. The material that have low absorption cross section Σ_a (barns) and high diffraction cross-section Σ_d (barns). The moderation efficiency can be determine based on the moderation power (barns) and moderation ratio using equation 2.6 and 2.7 [22].

$$\text{Moderation Power} = \xi \Sigma_d \quad (2.5)$$

To become a good moderator material, each material should have high moderation ratio. The equation 2.7 below show the diffraction cross-section Σ_d (barns) over absorption cross section Σ_a (barns).

$$\text{Moderation ratio} = \frac{\xi \Sigma_d}{\Sigma_a} \quad (2.6)$$

2.2. BNCT design and modifications

In this review, the arrangement, geometry and the design of the BNCT facility are being considered to optimize the thermal and epithermal neutron flux for successfully use as a beam in BNCT studies. There are several case studies that are being observed around the world in to order to find the feasibility in modifying the epithermal neutron and the thermal neutron at the research reactor for BNCT. The Table 1 shows the recommended value for BNCT by [11].

Table 1. The recommended value for BNCT by IAEA [11]

BNCT Beam Port Parameters	Recommended value
Thermal neutron	$>10^9$ neutron.cm ⁻² s ⁻¹
Gamma flux	$<10^{-13}$ γ cm ⁻² s ⁻¹

In Iran, the research regarding to BNCT has taken place at the Tehran research Reactor (TRR). TRR is a TRIGA type reactor, which has a thermal column with the cross section of 1.2 m x 1.2 m and 3 m length. Initially, this thermal column has been filled with the graphite blocks, but for the BNCT purpose, they need to modify the thermal column to meet the thermal neutron beam criteria specified by IAEA. Figure 4 shows the thermal column of TRR before the modifications while the Figure 5 shows the thermal column after the modifications.

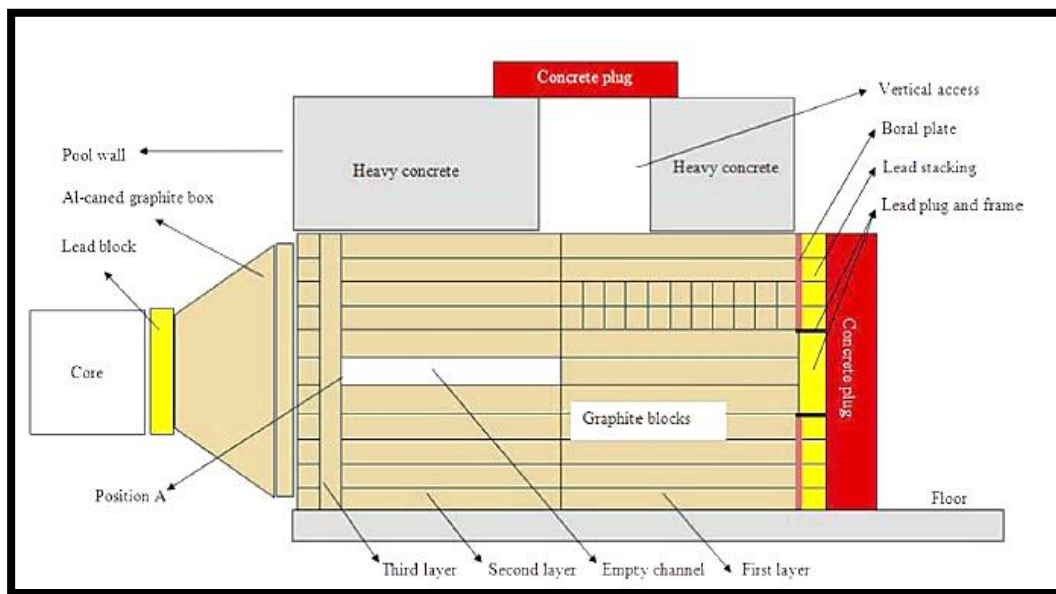


Figure 4. Schematic diagram of thermal column before modifications [23].

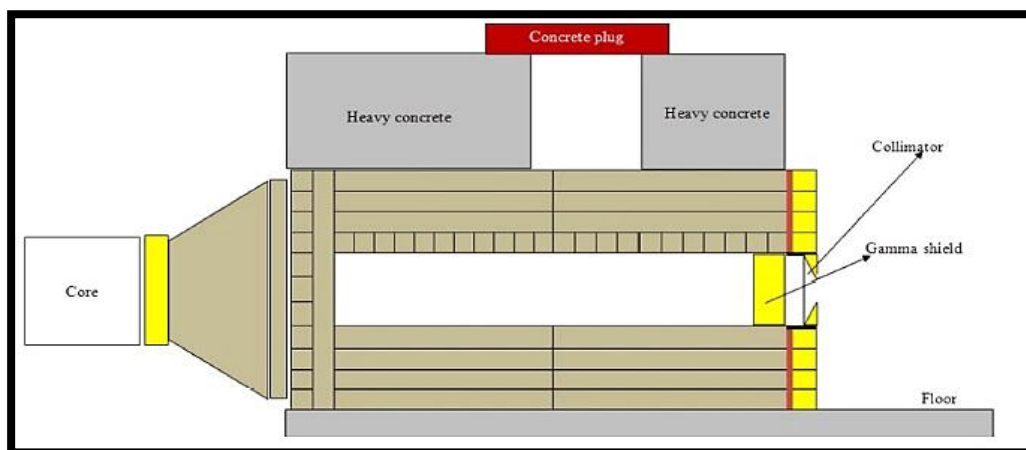


Figure 5. Schematic diagram of thermal column after modifications [23].

The main objective of the modification of the thermal column at the TRR is achieved by rearranging the graphite blocks as shown in the figure above. Besides that, the gamma dose rate at the exit of the thermal column is also considered by placing the gamma shield before the collimator. Based on the measurement using neutron activation analysis and TLD-700 dosimeter, the modification of the thermal column for BNCT purpose at TRR has produced about 5.6×10^8 neutron.cm⁻² s⁻¹ of the thermal neutron beam and the gamma dose rate calculated was 0.57 Gyh⁻¹ while the cadmium ratio (CR) is 186 for gold foils. The advantages of this modification are minimal changes in the thermal column structure thus there are no disruption to the other TRR normal activities. Another advantage is the sample and the phantom for BNCT can be irradiated outside of the thermal column [23]. At the same time, the TRR researchers also found the potential to produce the epithermal neutron at the TRR based on the determination using MCNP (Monte Carlo N-Particle). For further research, the BSA (beam shaping assembly) has been designed and also modified to produce the epithermal neutron flux intensity required by IAEA for BNCT purpose. At first to optimize the thermal column to produce epithermal neutron as recommended by IAEA, the BSA design had been made by using MCNP4 code by considering that all the graphite blocks had been

removed. The design of the BSA included the important parts such as thermal neutron filter, neutron reflector, neutron moderator and the gamma filter with the recommended of the suitable material use for BNCT by IAEA [23].

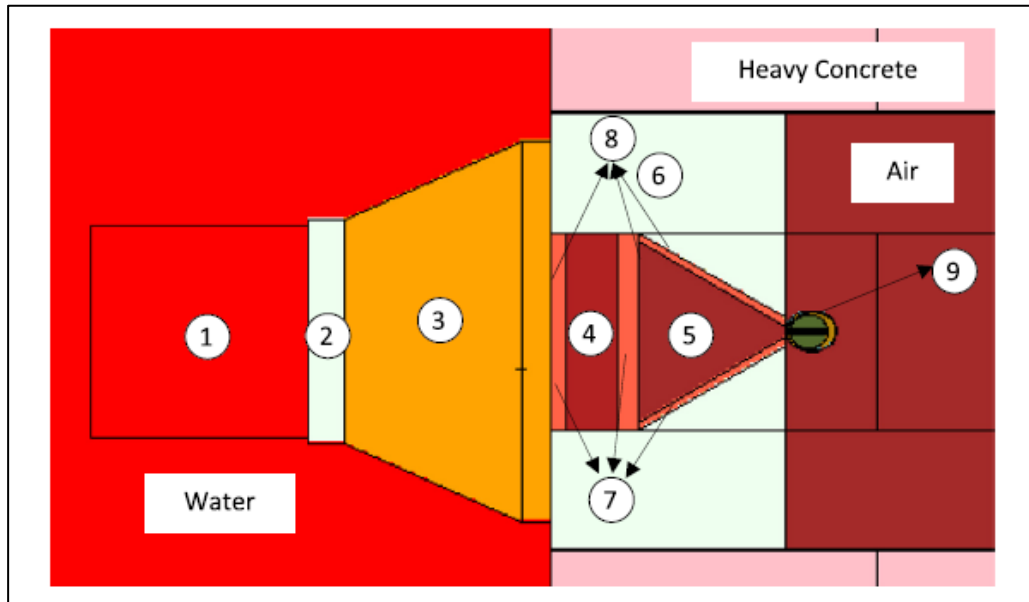


Figure 6. MCNP geometry of BSA for epithermal neutron beam BNCT [23]

As show on the Figure 6, labelled 1 on the design shows the reactor core. The reactor core must be considered as a neutron source for the reactor. The number 2 on the design is a gamma filter in which the material use is lead block. This material was used to sufficiently filter the prompt gamma produce from the reactor core. Number 3, is an Aluminium canned graphite blocks and number 4 is the neutron moderator. In this design, the neutron moderator become the neutron parameter to be observe and the material use where are Al, Al_2CO_3 , Al_3F , MgF_2 , PbF_2 , Al/AlF₃ alloy (30% Al, 70% AlF₃), and Flunatal alloy (30% Al, 69% AlF₃, 1% LiF) with different thickness. Number 5 on the BSA was the air that filled inside the collimator while the number 6 was the neutron reflector. The neutron reflector in this design is solid material in the form of Pb and PbF₂ with different thickness. The gamma filter was being label as 7 with the use of bismuth material and the number 8 show that the thermal neutron filters that use Cd (cadmium) at the beginning and the end of the neutron moderator with 2 mm thickness. Number 9 on the design shows the phantom or sample place [23].

The result of the BSA design using simulation of MCNP4 shows that the epithermal neutron produce was $> 1.0 \times 10^9 \text{ neutron.cm}^{-2} \text{ s}^{-1}$ and the fast neutron produce was $> 4 \times 10^8 \text{ neutron.cm}^{-2} \text{ s}^{-1}$ that had proved the material use in this design was feasible enough. The use of thermal neutron filter also helps to give the higher value of epithermal neutron flux produce at the beam port as the thermal neutron was absorb by the cadmium sheets. According to Kasesaz and friends [23], the BSA design can be suggested for the thermal column of TRR with the use of 20 cm of aluminium as the best neutron moderator and the 2 mm cadmium sheet as thermal neutron filter that can highly produce about $> 1.0 \times 10^9 \text{ neutron.cm}^{-2} \text{ s}^{-1}$ of epithermal neutron beam. As summarize, there are several materials that had studied their characteristics towards neutron by few researchers around the world for BNCT purpose. Neutron shielding material studied by a few researchers for BNCT purpose include material such as Bismuth, Flunatal, Aluminum, CF₂, and Heavy Water [24,25,26,27,3,28]. On other hand, materials such as Lithium Polyethylene, Polyethylene, Li₂CO₃-PE, Pb, Nickel and Paraffin have proven consist of neutron collimate characteristic [29,18,26,27,24,28]. Lastly, the study of material characteristic towards neutron show that material such as Aluminium, Heavy Concrete, Ordinary concrete, Borated Polyethylene, Polyethylene and Ice [24,25,30,31,28,5].

3. Current Status on BNCT At Triga Mark-II

3.1. Measurement of neutron flux at the thermal column of RTP

Initially, the measurement of the thermal neutron flux was conducted along the thermal column at 250 kW. The thermal column was divided into 3 phases (Phase 1, Phase 2 and Phase 3 as display in Figure 7) so that an accurate measurement can be obtained by using gold foil activation method. This value was used as a benchmark for the neutron flux produced from the thermal column for BNCT purpose [32].

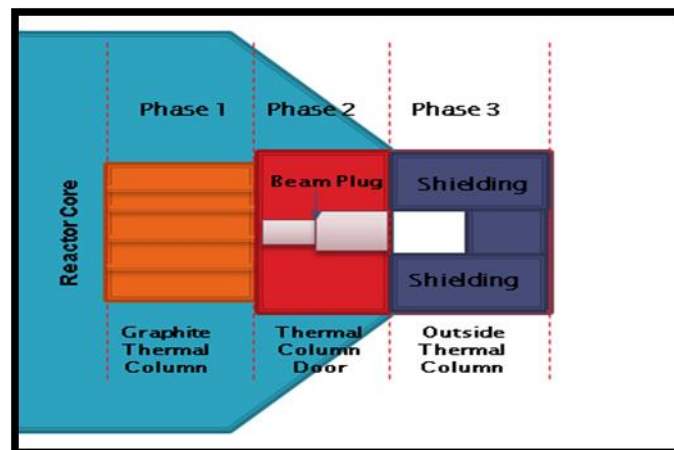


Figure 7. Phase 1, Phase 2 and Phase 3 at the thermal column of RTP [33].

In the Phase 1 measurement, there are two main data to be analysed which is thermal neutron flux and epithermal neutron flux. From the data tabulated in Table 4.1, stringer G7 produced the highest thermal neutron flux compared to E4, E10, L7, J10 and J4 with 7.42×10^{11} neutron.cm⁻².s⁻¹ at the sample position of G19. In contrast, the lowest thermal neutron flux produced from this measurement was from the stringer E10 at the sample position of E18. Based on Table 2, the positioning of the stringer was one of the main factors that contribute to the production of thermal neutron flux. The stringer G7 was located at the centre of the graphite thermal column and the closest stringer to the RTP core compared with E4, E10, L7, J10 and J4 [32]. In contrast, the position of stringer E10 (which is located above the reactor core) demonstrated the lowest thermal neutron flux due to the high volume of shielding material. Besides that, the epithermal neutron flux graph indicated that the highest result of epithermal neutron came from holes G19 with 1.16×10^{11} neutron.cm⁻².s⁻¹ compared to the lowest result from stringers L7. The location of stringer L7 was nearest to the RTP moderator and produced the effect of thermalization. The interaction of the irradiated samples and coolant as an effect of thermalization has produced less epithermal neutron flux. The thermalization also affects the measured value of cadmium ratio from L7.

The statistical analysis using the gold foil activation method can be used to determine the value of the cadmium ratio. The cadmium ratio is defined as the evaluation of the activating nuclide of epithermal neutron compared to the thermal neutron [33]. The measurement of the cadmium ratio is extensive in Phase 1 to determine the quality of the thermal and epithermal neutrons produced from the stringer. The holes L47 from stringer L7 significantly produced the highest cadmium ratio with 247.34 compared to other stringers. In contrast, most of the holes positioned next to the reactor core produced a lower cadmium ratio compared to the holes positioned farther from the reactor core. For example, holes G19 from stringer G7 have the lowest cadmium ratio due to their positioning. This concludes that stringer L7 had a high active nuclide of epithermal neutron compared to the thermal neutron despite having the lowest epithermal neutron flux and thermal neutron flux. To construct validity, the gold foil activation measurement at Phase 1, the experiment results from this research were compared with measurement results using the gold foil activation method from previous BNCT research at the thermal column at RTP [34]. Table 3 shows the comparison

between the epithermal neutron flux and thermal neutron flux by using foil activation measurement from this research and previous research.

Table 2. Phase 1 measurement of epithermal neutron flux and thermal neutron flux using the gold foil activation method [32].

Stringers	Holes	Thermal Neutron Flux, $\text{neutron.cm}^{-2}\text{s}^{-1}$	Epithermal Neutron Flux, $\text{neutron.cm}^{-2}\text{s}^{-1}$
E4	E07	7.05×10^{10}	1.11×10^{10}
	E09	2.34×10^{10}	3.10×10^8
E10	E12	2.24×10^{10}	2.18×10^9
	E13	5.75×10^{10}	2.64×10^9
	E16	1.04×10^{10}	1.33×10^9
	E18	1.35×10^9	6.61×10^7
G7	G19	7.42×10^{11}	1.16×10^{11}
	G24	2.81×10^{11}	2.30×10^9
	G26	2.41×10^{10}	1.32×10^8
	G28	1.47×10^{10}	2.03×10^8
	G31	1.85×10^{10}	1.73×10^8
J4	J32	5.39×10^{10}	2.94×10^9
	J35	8.31×10^{10}	2.12×10^8
	J37	7.31×10^{10}	1.81×10^8
J10	J38	6.78×10^{10}	2.50×10^9
	J40	1.63×10^{10}	2.07×10^8
	J43	9.79×10^9	1.13×10^8
L7	L44	2.10×10^{10}	1.24×10^9
	L46	1.48×10^{10}	6.65×10^7
	L47	1.20×10^{10}	3.98×10^7

Table 3. The measurement of thermal and epithermal by using gold foil activation method [32,34].

Position	This research, $\text{neutron.cm}^{-2}\text{s}^{-1}$		Munem (2007), $\text{neutron.cm}^{-2}\text{s}^{-1}$	
	Thermal	Epithermal	Thermal	Epithermal
G19	7.42×10^{11}	1.16×10^{11}	2.64×10^{11}	7.63×10^9
G24	2.81×10^{11}	2.30×10^9	2.45×10^{10}	5.79×10^7
G28	1.47×10^{10}	2.03×10^8	3.75×10^9	4.58×10^6
G31	1.85×10^{10}	1.73×10^8	3.11×10^8	7.47×10^5

Based on Table 3, the data of the neutron flux measured by both researches from position G19, G24, G28 and G31 follows the decrease trend but there is some difference in terms of flux. The differences between reactor power operation during the measurement had affected the flux production for both epithermal neutron and thermal neutron. Thermal neutron produces were within the range of 10^{11} to 10^{10} $\text{neutron.cm}^{-2}\text{s}^{-1}$ for 250 kW [32]. The thermal neutron flux was flutter due to the reactor power noise on the neutron distribution that had effect on random neutron fluctuation [35]. The epithermal neutron flux and thermal neutron flux of Phase 3 and Phase 2 has been displayed on the Table 4.4. The result shows that the stringers G7 produce the maximum production of thermal neutron flux of 3.62×10^7 $\text{neutron.cm}^{-2}\text{s}^{-1}$ at the distance of 142 cm. Basically, the neutron flux decreases linearly with the length. Thus, at the Phase 2, both thermal and epithermal neutron flux declining from the 10^7 to the 10^6 for thermal neutron. For the Phase 3, the results of the thermal neutron flux decline about 25% from the 3.53×10^6 $\text{neutron.cm}^{-2}\text{s}^{-1}$

$^2\text{s}^{-1}$ and flutter back to $1.28 \times 10^6 \text{ neutron.cm}^{-2}\text{s}^{-1}$ at the last sample (A6) in the distance of 283.9 cm from the thermal column. The fluctuation of the thermal neutron was occurred when there was air exposure along the Phase 3 and the interaction between them.

The measurement of epithermal neutron flux also showed the flux decline along the distance. At the Phase 2, the initial flux of epithermal neutron was $5.91 \times 10^6 \text{ neutron.cm}^{-2}\text{s}^{-1}$ at the distance 142 cm from the graphite thermal column and continuously decrease to the $8.90 \times 10^5 \text{ neutron.cm}^{-2}\text{s}^{-1}$ at the end of Phase 2. At the Phase 3, the last sample A6 produced only $3.89 \times 10^4 \text{ neutron.cm}^{-2}\text{s}^{-1}$ which is about 99.34% reduces from the initial epithermal neutron flux from Phase 2. Technically, the epithermal neutron shows the gradually decline of flux along Phase 2 and Phase 3 without any fluctuation and follow the standard law of radiation and distance [32]. Both neutron thermal and epithermal neutrons flux decrease with the distance from 142 cm to 283.9 cm from graphite thermal column as the flux generally is exponentially decreasing with distance [36]. Based on Figure 8, from the Phase 1 to the Phase 2, the thermal and epithermal neutron flux is significantly reduced about 99.8% for thermal neutron flux and 96.5% for epithermal neutron flux. The uses of graphite as a neutron moderator for the research reactor in Phase 1 become the main factor for the tremendously decreases of flux from the Phase 1 to the Phase 2. The properties of graphite that have higher thermal conductivity have increased the measurement of thermal and epithermal neutron flux at Phase 1 [37]. Across the beam line, the thermal neutron flux was higher than epithermal neutron flux with $1.00 \times 10^2 \text{ neutron.cm}^{-2}\text{s}^{-1}$ difference because the thermal column itself produce are well known as the thermal neutron source at the RTP [38]. The aim of this measurement to clearly see the thermal and epithermal neutron across the beam line for the optimization purpose for the collimator [32].

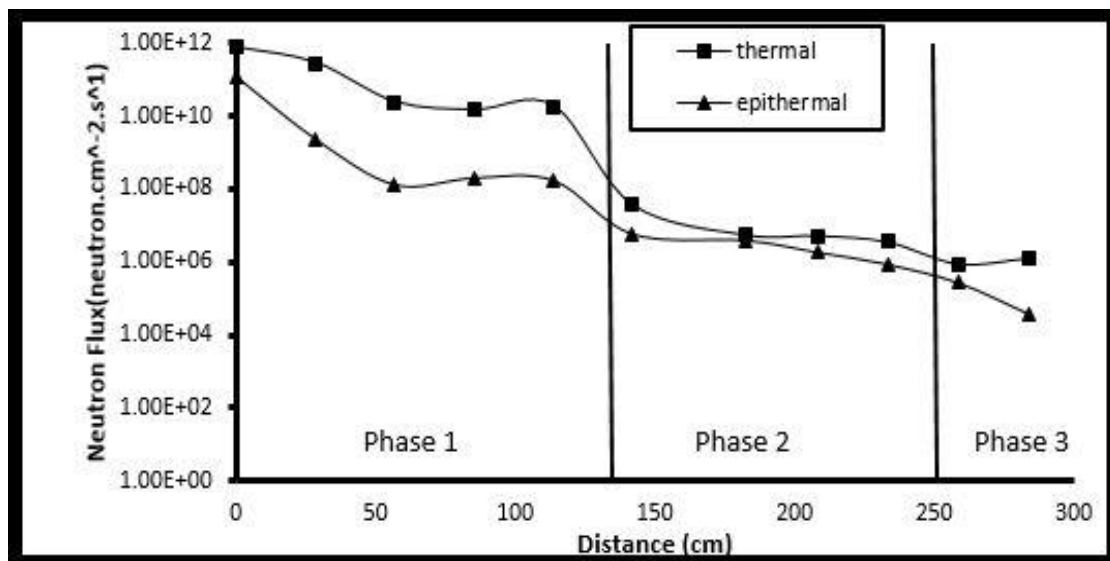


Figure 8. The neutron flux measurement across the beam line for the Phase1, Phase 2 and Phase 3 [32].

3.2. Characterization of materials for BNCT collimator

The characterization of material for BNCT collimator were done by experimental approach (TLD and Microspec-6) at the thermal column. The characterization of material was important to ensure the material use for collimator can optimize the thermal neutron beam and reduces other ionising particles. 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 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 as shown in Figure 9 for neutron.

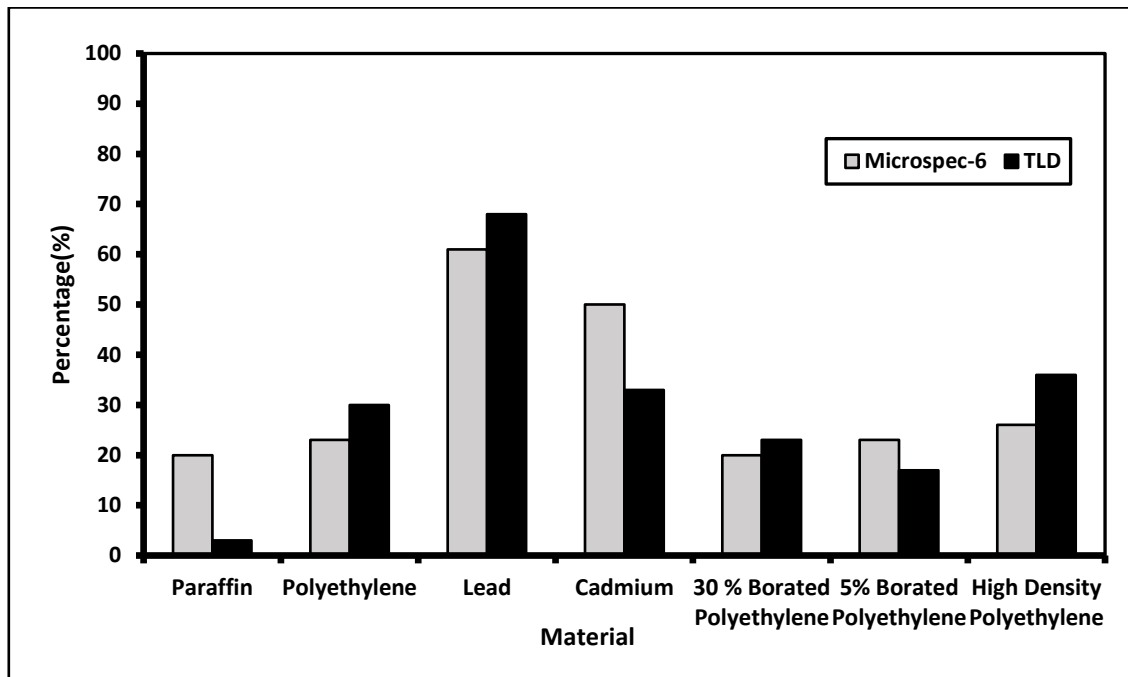


Figure 9. The percentage of neutron produce by using Microspec-6 and TLD detector.

Based on Figure 9 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 detector with 17%. Previous research conclude that the material paraffin can be used as a neutron moderator and collimate for BNCT based on the research by using Microspec-2 [38]. 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% [39]. Besides that, according to Devine, 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 [40]. The measurement by using both Microspec-6 and TLD demonstrated the same pattern. According to result and discussion from both experiment, paraffin and 30% borated polyethylene were chosen 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 [41]. 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 [42].

3.3. Design of BNCT collimator

The neutron collimator for BNCT research facility at RTP is divided into two different position of collimator which is internal collimator (Phase 2) and external collimator (Phase 3). The internal collimator located at the Phase 2 that already has specific dimensions while the external collimator located at the Phase 3 which is before the sample. The internal collimator function to focus the neutron beam in towards the exit of the collimator while moderating fast neutrons and to filter and absorb gamma radiation while the external collimator function is to channel the neutron beam towards the exit for irradiating sample, while absorbing scattered neutrons and gamma. The designing collimator for research reactor purpose by doing measuring directly and without simulation will lead to radiation accident [43]. In fact, the Monte Carlo method to design the collimator was already integrated within the MCNPX and can accurately models the neutron transport phenomena [44]. As mentioned before, the internal collimator located at the Phase 2 of the thermal column has a specific size and dimensions. In the internal collimator, there are two parts which are inner part and outer part. Both parts have different diameter and length, as the inner part has small diameter which is 20 cm with 51.8 cm long while the outer part consist of 24 cm diameter and 54.8 cm of length. In fact, the collimators work based on the principle of radiation interaction with matter, by shifting or filtering methods. The shifting method uses a moderator to lower the neutron energy whereas the filtering method use materials to absorb neutrons in certain energies [45].

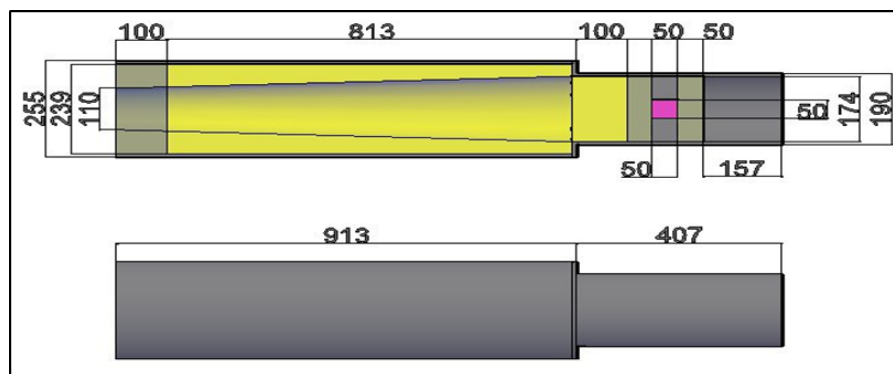


Figure 10. Collimator design using simulation (The number represent the dimension in unit millimetre (mm)).

Basically, the Figure 10 shows the design of the collimator based on the filtering method at the inner part and shifting method at the outer part of the collimator by using simulation software of MCNPX. The filtering method was described on the 5 cm of hollow cylinder arrange along the inner part of the collimator. Meanwhile, the shifting method was described by the conical shape towards the end of the collimator. In terms of the collimator casing, the case was fixed and made up by the aluminium grade 5754 with 4 mm thickness. Both inner part and outer part of the collimator use aluminium 5754 materials. The aluminium 5754 functions as the internal collimator case. The main reason of using aluminium grade 5754 because of its characteristic that have a low neutron activation element [42]. The aluminium 5000 series also are extensively used in the research reactor because of their good corrosion resistance in the deionized water under 80 Celsius and the good mechanical properties at the temperature under 150 Celsius. The internal collimator has been used to put any neutron collimator, shielding, filter and moderator material for collimator purpose to optimize the beam of thermal neutron produce from the thermal column for BNCT purpose [46]. The result of collimator design based on material characterization was compared with preliminary measurement without collimator in Figure 11 [32,47]. Figure 11 display the thermal neutron flux along the thermal column with and without collimator on Phase 2. The presence of collimator at Phase 2 of thermal column had better thermal neutron production rather than without the collimator install. At the distance 142 cm or at the initial of the Phase 2, the experiment show that the thermal neutron flux was drop from the 10^{10} neutron.cm^{-2s-1} to 10^7 neutron.cm^{-2s-1}.

$2s^{-1}$ but with the presence of collimator, the thermal neutron flux was slightly lower within the range of 10^8 neutron.cm $^{-2}s^{-1}$. The collimator was sustained the thermal neutron flux until at the 283.9 cm with only 20% of thermal neutron reduced. On other hands, from the Phase 2 to the Phase 3 without the collimator an experiment was measured that the thermal neutron was reduced to 96%. Overall, the thermal neutron with the collimator installation can sustain the higher thermal neutron flux from the RTP that can be used for BNCT research [47].

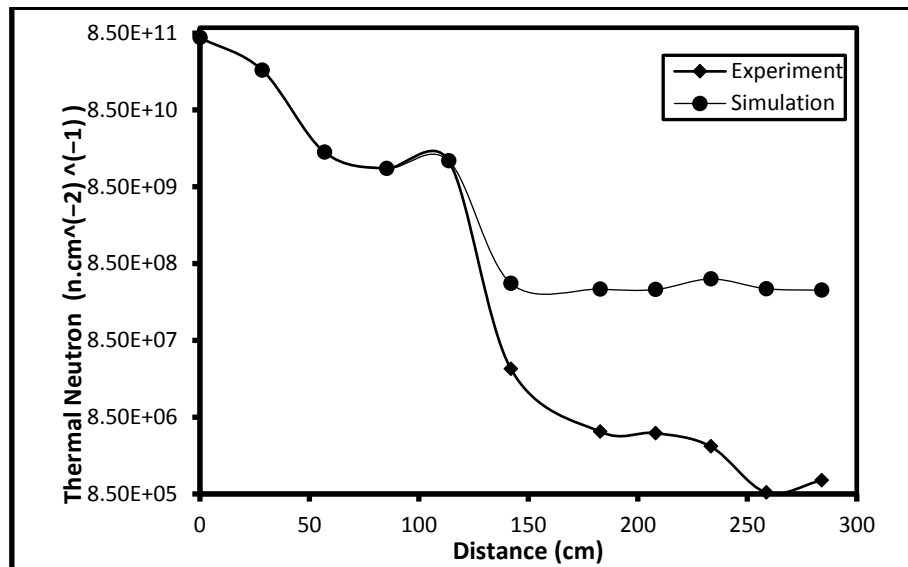


Figure 11. The graphical different of thermal neutron flux along the thermal column by simulation (collimator) and experiment [32,47].

4. Future research

The complete BNCT facilities should have biological shielding to be used during the research and experiment. In clinical and research of BNCT, the shielding was one of the important aspects to be overcome before the future research of BNCT can carried out. The shielding must be good enough to shield neutron and secondary gamma that would affect the BNCT research. Therefore, the future research on shielding can be carried out as extended from this review. Besides that, the external collimator must be included to optimize the beam for BNCT research and to minimize the gamma from the centre of collimator. The external variable must have variable aperture for the purpose of BNCT research on different cancer in future.

5. Conclusion

There is only a research reactor in Malaysia that was identified as an available neutron source for BNCT. The thermal column of RTP is determined to be a suitable place for BNCT purpose based on the early studied of BNCT. The measurement with the verification by experimental and simulation for the thermal neutron at the thermal column was precise that the availability of the thermal column of Malaysia RTP was sufficient enough for the development of BNCT as the quality and intensity of the neutron beam produced was well within a standard requirement for BNCT facility. The review on the material for BNCT collimator around the world benefit on specific material use for designing the ideal collimator and beam optimization. The material for designing collimator divide into neutron shielding materials, neutron moderator materials and neutron collimate material. This review specifically characterizes those material based on their characteristic. This paper also reviews the current status of BNCT at the RTP with the measurement of the thermal neutron flux was conducted along the thermal column at 250 kW.

The thermal column of RTP was divided into 3 phases (Phase 1, Phase 2 and Phase 3) so that an accurate measurement can be obtained by using gold foil activation method. This value in the range of 1.0×10^6 neutron.cm⁻²s⁻¹ as a benchmark for the neutron flux produced from the thermal column. The collimator was designed using different types of materials, and their characteristic towards gamma and neutron flux was investigated. The reviewed demonstrated that the final thermal neutron flux produced was significantly for BNCT purpose with 10^8 neutron.cm⁻²s⁻¹ with collimator design. Ultimately, this review recommends the future research can be conducted on BNCT at RTP. Thus, the aim of this paper is to review the thermal neutron beam by designing the external neutron collimator for the BNCT purpose so that the neutron flux obtained is within the range that set by the IAEA standard in the future.

Acknowledgements

Authors would like to extend their gratitude to Ministry of Higher Education Malaysia and Universiti Malaysia Pahang (UMP) with grant number RDU192615.

Reference

- [1] Bray F, Ferlay J, Soerjomataram I, Siegel RL, Torre LA, Jemal A 2018 Global cancer statistics 2018 GLOBOCAN estimates of incidence and mortality worldwide for 36 cancers in 185 countries *CA Cancer J Clin* **68**(6):394–424.
- [2] WHO 2003 The World Health Report 2003- Shaping the Future. Geneva: Switzerland.
- [3] Durisi, E, Zanini, A, Manfredotti, C, Palamara, F, Sarotto, M, Visca, L, & Nastasi, U. 2007 Design of an epithermal column for BNCT based on D–D fusion neutron facility *Nuclear Instruments and Methods in Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipment* **574**(2) p 363-369.
- [4] Faião-Flores F, Coelho PRP, Muniz ROR, Souza GS, Arruda-Neto J, Maria DA 2011 Antitumor potential induction and free radicals production in melanoma cells by Boron Neutron Capture Therapy *Appl. Radiat. Isot.* **69**(12):1748–51.
- [5] M. Solleh MR. 2016 Moderator, Collimator and Shielding Studies for BNCT Research at Malaysian Nuclear Agency (Universiti Sains Malaysia: Phd Thesis).
- [6] Kraft G. 2000 Tumor therapy with heavy charged particles *Prog. Part. Nucl. Phys* **45**:S473–544.
- [7] Valda A, Minsky DM, Kreiner AJ, Burlon AA, Somacal H 2005 Development of a tomographic system for online dose measurements in BNCT (Boron Neutron Capture Therapy). *Brazilian Phys.* **35**:785–8.
- [8] Gohil CJ, Noolvi MN 2015 Selective cancer treatment by Boron Neutron Capture Therapy (BNCT)-a review *Int. J. Pharm. Chem. Anal* **2**(3):136–8.
- [9] Nigg D 2019 Neutron Sources and Applications in Radiotherapy A Brief History, and Current Trends.
- [10] Chen SN Neutrons. In: Encyclopedia of Cancer and Society [Internet]. 2455 Teller Road, Thousand Oaks California 91320 United States: SAGE Publications, Inc.
- [11] International Atomic Energy Agency 2001 Current status of neutron capture therapy (IAEA: TECDOC-1223).
- [12] Bosko A, Zhilchenkov D, Reece W 2004 GE PETtrace cyclotron as a neutron source for boron neutron capture therapy *Appl. Radiat. Isot.* **61**(5):1057–62.
- [13] Kankaanranta L, Seppälä T, Koivunoro H, Saarilahti K, Atula T, Collan J 2007 Boron Neutron Capture Therapy in the Treatment of Locally Recurred Head and Neck Cancer *Int. J Radiat. Oncol* **69**(2):475–82.
- [14] Bilski P, Obryk B. 2006. Thermoluminescent dosimetry at the IFJ Krakow. *Properties of Semiconductor Nanostructures (Electronic Materials vol 4)*
- [15] Joensuu H, Kankaanranta L, Seppälä T, Auterinen I, Kallio M, Kulvik M 2003 Boron neutron capture therapy of brain tumors: clinical trials at the Finnish facility using boronophenylalanine *J. Neurooncol* **62**(1–2):123–34.
- [16] Pohjola M 2009 Biologically targeted treatment.
- [17] Beddoe AH 1997 Boron neutron capture therapy *Br. J. Radiol* **70**(835):665–7.

- [18] Akan, Zafer, Türkmen, Mehmet, Çakır, Tahir, Reyhancan, İskender A, Çolak, Üner, Okka, Muhittin, & Kızıldağ, Sahip 2015 Modification of the radial beam port of ITU TRIGA Mark II research reactor for BNCT applications *Applied Radiation and Isotopes* p110-116.
- [19] Kreiner AJ, Bergueiro J, Cartelli D, Baldo M, Castell W, Asoia JG 2016 Present status of Accelerator-Based BNCT *Reports Pract. Oncol. Radiother* **21(2)**:95–101.
- [20] Allen BJ, Ralston A. 1997. *Boron dose enhancement for ²⁵²Cf Brachytherapy*. In: Larsson B, Crawford J, Weinreich R, editors. *Advances in Neutron Capture Therapy: Volume I, Medicine and Physics*. London: Elsevier Health Sciences.
- [21] Cacuci DG 2010 *Handbook of Nuclear Engineering* [Internet]. Boston, MA: Springer
- [22] Lamarsh JR, Baratta AJ 2001 *Introduction to Nuclear Engineering 3rd ed* Pearson.
- [23] Kasesaz Y, Khalafi H, Rahmani F, Ezzati A, Keyvani M, Hossnirokh A 2014 Design and construction of a thermal neutron beam for BNCT at Tehran Research Reactor *Appl. Radiat. Isot* **94** p149–51.
- [24] Fauziah, Nina, Widiharto, Andang, & Sardjono, Yohannes 2015 A Conceptual Design Of Neutron Collimator In The Thermal Column Of Kartini Research Reactor For In Vitro And In Vivo Test Of Boron Neutron Capture Therapy *Tri Dasa Mega-Jurnal Teknologi Reaktor* **15(2)**.
- [25] Hawk, Andrew E, Blue, TE, & Woollard, JE 2004 A shielding design for an accelerator-based neutron source for boron neutron capture therapy *Applied radiation and isotopes* **61(5)** p1027-1031.
- [26] Shaaban, Ismail, & Albarhoum, Mohamad 2015 Design calculation of an epithermal neutronic beam for BNCT at the Syrian MNSR using the MCNP4C code *Progress in Nuclear Energy* **78** p297-302.
- [27] Monshizadeh, M, Kasesaz, Y, Khalafi, H, & Hamidi, S 2015 MCNP design of thermal and epithermal neutron beam for BNCT at the Isfahan MNSR. *Progress in Nuclear Energy* **83** p427-432.
- [28] Sarif MA 2016 Study of Moderator (Collimator) Materials for Boron Neutron Capture Therapy (BNCT) Facilities at Thermal Column RTP.
- [29] Faghihi F, Khalili S 2013 Beam shaping assembly of a D–T neutron source for BNCT and its dosimetry simulation in deeply-seated tumor *Radiat. Phys. Chem* **89:1**–13.
- [30] Pouryavi M, Farhad Masoudi S, Rahmani F 2015 Radiation shielding design of BNCT treatment room for D-T neutron source *Appl. Radiat. Isot.* **99:90**–6.
- [31] Sakurai Y, Sasaki A, Kobayashi T 2004 Development of neutron shielding material using metathesis-polymer matrix *Nucl. Instruments. Methods Phys. Res. Sect. A. Accel. Spectrometers, Detect. Assoc. Equip.* **522(3)**:455–61.
- [32] Munem, Eid Mahmoud Eid Abdel 2007 Neutron flux measurements with Monte Carlo verification at the thermal column of a Triga mark II reactor: Feasibility study for a BNCT facility (Universiti Sains Malaysia: Phd Thesis).
- [33] Shalbi, S., Salleh, W. N. W., Idris, F. M., Rosdi, M. A. A., Sarkawi, M. S., Jamsari, N. L., & Nasir, N. A. M 2018 Neutron measurement at the thermal column of the Malaysian Triga Mark II reactor using gold foil activation method and TLD *In IOP Conference Series: Materials Science and Engineering* (Vol. **298**, No. 1, p. 012031). IOP Publishing.
- [34] Abdel Munem, E., Shukri, A., & Tajuddin, A. A 2006 Neutron flux measurement in the thermal column of the Malaysian TRIGA mark II reactor with MCNP verification.
- [35] Waheed, A., Ali, N., Baloch, M. A., Qureshi, A. A., Munem, E. A., Rajput, M. U and Muhammad, W 2017 Optimization of moderator assembly for neutron flux measurement: experimental and theoretical approaches *Nuclear Science and Techniques* **28(5)** 61.
- [36] Rosdi, M. A. A., Goh, P. S., Shalbi, S., Idris, F., Jamro, R., & Isa, N. M 2019 Monte Carlo simulation for designing collimator of neutron diffractometer facility in Malaysia *In IOP Conference Series: Materials Science and Engineering* (Vol. 555, No. 1, p. 012016).
- [37] Marashdeh, M. W., Al-Hamarneh, I. F., Munem, E. M. A., Tajuddin, A. A., Ariffin, A., & Al-Omari, S 2015 Determining the mass attenuation coefficient, effective atomic number, and electron density of raw wood and binderless particleboards of *Rhizophora* spp. by using Monte

- Carlo simulation. Results in Physics **5** p228-234.
- [38] Arif MA 2016 Study of Moderator (Collimator) Materials for Boron Neutron Capture Therapy (BNCT) Facilities at Thermal Column RTP.
- [39] Paiva F, Siqueira PTD, Cavalieri TA 2015 Comparing the responses of TLD 100, TLD 600, TLD 700 and TLD 400 in mixed neutron-gamma fields. In: *INAC 2015: International Nuclear Atlantic Conference Brazilian Nuclear Program State Policy For A Sustainable World*.
- [40] Devine RT, Romero LL, Gray DW, Seagraves DT, Olsher RH, Johnson JP 2002 Evaluation of spectrum measurement devices for operational use. *Nucl. Instruments. Methods Phys. Res. A. Accel. Spectrometers, Detect. Assoc. Equip.* **476(1–2)**:416–22.
- [41] Korkut T, Karabulut A, Budak G, Aygün B, Gencil O, Hançerlioğulları A 2012 Investigation of neutron shielding properties depending on number of boron atoms for colemanite, ulexite and tincal ores by experiments and FLUKA Monte Carlo simulations *Appl. Radiat. Isot.* **70(1)**:341–5.
- [42] Shalbi, S., Salleh, W. N. W., Rosdi, M. A. A., & Idris, F. M 2019 Optimization and designing collimator for boron neutron capture therapy at the thermal column of RTP *IOP Conference Series: Materials Science and Engineering* (Vol. **555**, No. 1, p. 012017).
- [43] Payudan, A., Haryadi, A., & Abdullatif, F 2017 Optimization of Collimator Neutron Design for Boron Neutron-Capture Cancer Therapy (BNCT) Based Cyclotron 30 MeV *Indonesian Journal of Physics and Nuclear Applications* **2(3)** 128-136.
- [44] Geng, C., Tang, X., Guan, F., Johns, J., Vasudevan, L., Gong, and Chen, D 2015 GEANT4 calculations of neutron dose in radiation protection using a homogeneous phantom and a Chin China hybrid male phantom *Radiation protection dosimetry* **168(4)** 433-440.
- [45] Ibrahim, T. N. S 2004 National Cancer Control Programme. Malaysia's Health 2004. Ministry of Health Malaysia.
- [46] Kapusta, B, Sainte-Catherine, C, Averty, X, Scibetta, M, & Rommens, M 2003 Present status on the mechanical characterization of aluminum alloys 5754-NET-O and 6061-T6 irradiated at high fluences *9th Meeting of the International Group on Research Reactors IGORR*.