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Dose Calculation and Measurement from $B^{10}(n, \alpha)Li^7$ Reaction Using Filtered Neutron Beam at Nuclear Research Institute

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Abstract: In this research, dose calculation and measurement from B^{10} (n, α) Li^7 reaction using filtered neutron beam at the Nuclear Research Institute have been reported. Calculation was carried out by Monte Carlo method using MCNP5 code. Neutron activation technique using vanadium foil was employed to determine neutron flux at various positions in phantom from which neutron dose has been calculated using conversion factor. These calculations are basics for the dose determination research of the Boron Neutron Capture Therapy (BNCT) in Vietnam.

Keywords: *Dose calculation, Monte Carlo, BNCT.*

I. INTRODUCTION

Boron neutron capture therapy (BNCT) is a promising treatment for malignant tumors and has been studied in many advanced countries. There are ongoing researches worldwide relating to the utilization of neutron beam in BNCT [1-3].

In Vietnam, however, BNCT has still been a new field. The only 500 kW nuclear reactor in Vietnam cannot generate sufficient epithermal neutrons for the BNCT; however, thermal neutrons can be obtained by using filtering system. These thermal neutrons can be used for basic experiments in the BNCT before real trials on living creatures. Besides, due to the lack of research facilities in Vietnam, simulation programs like MCNP will be valuable tools in supporting the studies. The combination of real experiments with simulation will be the alternative method to the development of the BCNT in Vietnam.

The main purpose of this research was to calculate radiation dose in the BNCT for brain tumors treatment using the simulation program of MCNP. Besides, the feasibility of MCNP program in simulating the neutron beam utilized at the Nuclear Research Institute (NRI) in Vietnam was also taken into consideration**.**

II. CONTENT

A. Subjects and methods

Neutron beam guide and neutron filter at Channel 2 of Dalat nuclear reactor modeled in MCNP were illustrated in Figure 1. The total length of the cylindrical-shaped neutron beam guide was 153 cm; inner diameter was 9.4 cm; outer rim consisted of 2 parts linked firmly together. The outer surface of the neutron beam guide was wrapped by a 4 mm layer of Aluminum. The inner bottom was a 3cm thick aluminum annulus to enable easy installation. The outer surface of the annulus was threaded. Its inner and outer diameters were 15 cm and 6.5 cm, respectively. The inner bottom of the annulus not only guided the neutron but also sealed the inner and outer surface, preventing the filter's pin and collimator from sliding out of the beam guide system. The outer bottom is an aluminum ring with a thickness of 2.7 cm,

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outer diameter of 15 cm, and inner diameter of 9.4 cm. The filter includes Bismuth and Silicon layers, with their thickness of 3 and 20 cm, respectively. After traversing the filter with maximum length of 150 cm, neutron beam was collimated. The collimator involved layers of materials, including: 3 layers of Lead with a total length of 30 cm, 5 layers of Borated + Hydrogenated Concrete (SWX-277 contained 1.56% B) with a total length of 60 cm. These layers were set alternately together. At 30cm away from the output gate of Channel 2, a block of 7cm of stainless steel was added to stop gamma radiation and ensure the watertight of the channel [4].

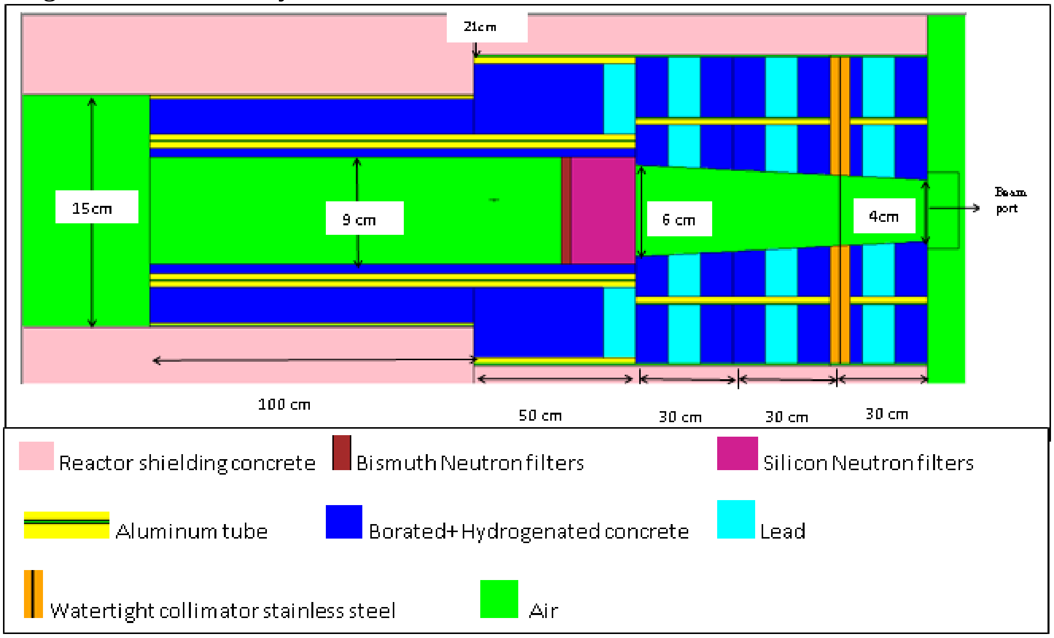
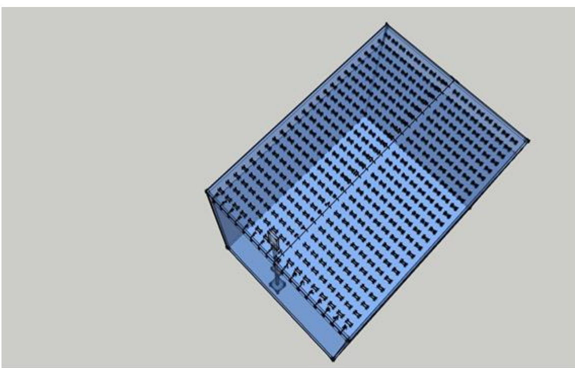


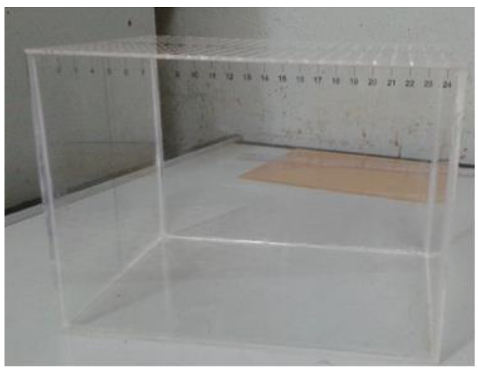
Fig. 1. Structure of the neutron beam guide and neutron filter simulated by MCNP5.

Requirements for the neutron beam guide of Channel 2: (i) Gamma and neutron dose rate outside of the beam guide < 10 μSv/h; (ii) Thermal neutron flux at the beam port $\geq 10^6$ n/cm^2/s; (iii) The output neutron flux had a cross section diameter of 3cm; (iv) Reduction of shielding outside the Channel and (v) Easy assembly and disassembly mechanism.

In this research, the neutron flux was determined at points distributed inside a water phantom. This phantom was designed as a rectangular plastic box, with its length, width and depth being 25 cm, 18 cm and 16 cm, respectively. The upper surface of the phantom was punctured with holes in an array form as shown in Figure 2.



(a)



(b)

Fig. 2. (a) Phantom simulation, (b) The real phantom.

Neutron flux was determined by the activation method. The standard samples used in this experiment were the round-shaped Vanadium (Vu - Vt) foils with neutron captured cross section of 4.75 barns, thickness of 0.05 mm. The foil was attached to a 16cm long stick which was inserted to the phantom through holes on the top surface as illustrated in Figure 2. The bottom of the stick was attached with a lead cube to keep the foil stable during the irradiation process. The irradiation time was nearly 4 minutes. Afterwards, the foil was removed from the phantom and measured by an HPGe detector. The neutron flux was then calculated from the radioactivity of the irradiated foil. The process was repeated with different positions of the foil within the phantom. Based on international standards of dose conversion coefficient for neutron flux, the dose rate can be determined from neutron flux measurements' results by multiplying by the elemental neutron kerma (Gy.cm^2) for ICRU adult brain. Data for adult brain component and neutron cross-section were extracted from ICRU Report 63 [5] and JENDL-3.2, respectively.

Table I. Elemental Neutron Kerma (Gy.cm^2) for ICRU Adult Brain.

Neutron Energy (MeV)	H^1	C^{12}	N^{14}	O^{16}	P^{31}	S-Nat	Total
1,000E-10	7,163E-14	5,514E-17	2,746E-12	3,806E-18	2,220E-17	8,709E-17	2,848E-12
2,530E-08	4,503E-15	3,467E-18	1,726E-13	2,393E-19	1,396E-18	8,709E-17	1,791E-13
1,100E-07	2,193E-15	1,762E-18	8,370E-14	2,982E-19	6,640E-19	8,709E-17	8,689E-14
1,100E-06	8,044E-16	1,392E-18	2,651E-14	2,073E-18	2,120E-19	8,709E-17	2,769E-14
1,100E-05	1,369E-15	8,754E-18	8,370E-15	2,045E-17	1,064E-19	8,709E-17	9,933E-15
1,100E-04	1,155E-14	8,596E-17	2,651E-15	2,045E-16	4,080E-19	8,711E-17	1,459E-14
1,100E-03	1,144E-13	8,567E-16	1,027E-15	2,045E-15	3,492E-18	8,727E-17	1,184E-13
1,100E-02	1,069E-12	8,481E-15	1,906E-15	2,037E-14	4,200E-17	8,883E-17	1,100E-12
1,050E-01	6,739E-12	7,493E-14	8,502E-15	1,903E-13	2,232E-16	1,043E-16	7,013E-12
1,050E+00	2,235E-11	4,068E-13	4,535E-14	2,776E-12	1,488E-15	3,349E-16	2,558E-11
1,050E+01	4,899E-11	1,704E-12	4,031E-13	9,728E-12	3,668E-14	1,249E-14	6,089E-11

B. Result and discussion

The total neutron flux at Channel 2 was 3.16×10^7 (n/cm^2.s), mainly consisting of thermal neutrons [4]. The measured and simulated thermal neutron flux distributions along the beam port's axis were given in Figure 3. As can be seen the Figure, there is a little difference between simulation and experiment results. This can be explained by the fact that in simulation, the flux in small cells was calculated instead of the flux at points. Therefore, the difference between simulation and measurement results was insignificant and the simulation of neutron beam generated from Channel 2 proved to be valid.

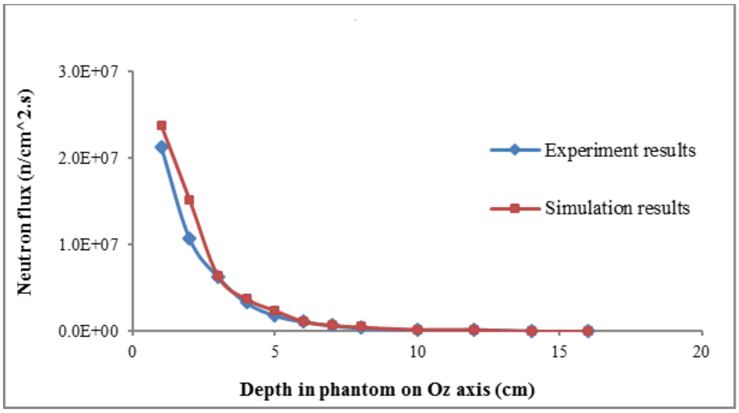


Fig. 3. Experiment and simulation results of neutron flux on Oz axis.

The experimental thermal flux distribution within the water phantom in the Oxz plane was also obtained and illustrated in Figure 4. From the Figure, significant neutron irradiation is observed within 1.5cm from the phantom surface. The isoflux area within the $2x10^7$-$2.5x10^7$ region had a maximum width of just a few millimeters, insignificantly by comparing to the radius of the beam port (2 cm). Once entering the phantom, neutrons scattered leading to the widening of neutron beam in water medium. Still, the thermal neutron flux declined dramatically with an increase in depth in the phantom.

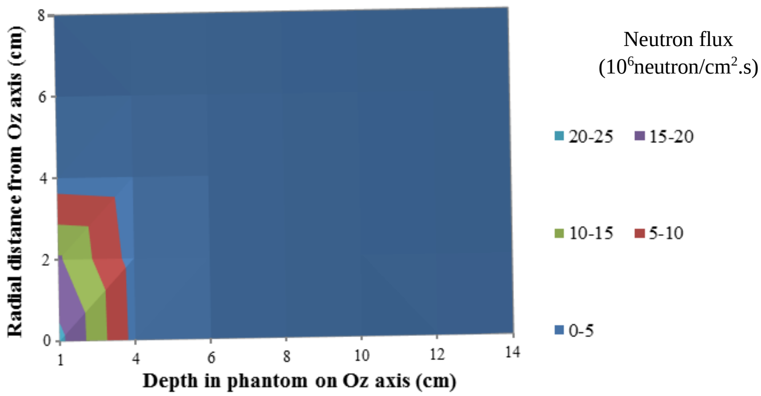


Fig. 4. Thermal neutron flux distribution within the water phantom on the Oxz plane.

Base on neutron flux determined by the activation method, radiation dose can be calculated. At each point of interest in the patient, one can identify four components contributing to the absorbed dose: (i) The gamma dose due to gamma rays accompanying the neutron beam as well as gamma rays induced when tissue absorbs thermal neutrons in $^1H(n,\gamma)$ ^2H reactions and emits 2.2 MeV gamma rays; (ii) The neutron dose caused by recoil protons from hydrogen in tissue in $^1H(n,n')p$ reactions; (iii) The proton dose resulting from locally deposited energy from the energetic proton and the recoiling ^{14}C nucleus when ^{14}N in tissue absorbs a thermal neutron and emits a proton in a $^{14}N(n,p)^{14}C$ reaction; (iv) The boron dose due to ^{10}B absorbs a thermal neutron in a $^{10}B(n,\gamma)^{11}B$ reaction. Total biological weighted effective dose and its component are calculated from the neutron flux as follows [6]:

Boron Dose:

$$D_B(Gy) = \phi_{th}.\frac{\sqrt{\pi}}{2}.\sigma_a^{B-10}(E_0).\frac{\frac{(w_{B-10})}{100}.N_A}{W_{B-10}}.Q \qquad (1)$$

Gamma Dose:

$$D_\gamma = \phi_{th}.N_H.\sigma_H.Q.f \times 1.6\times 10^{-13} \qquad (2)$$

Neutron Dose:

$$D_n = \phi_n.N_H.\sigma_H.E_n.f \times 1.6\times 10^{-13} \qquad (3)$$

Proton Dose:

$$D_p = \phi_{th}.N_N.\sigma_N.Q\times 1.6\times 10^{-13} \qquad (4)$$

Weighted Biological Effective Dose:

$$D_{bw} = w_c.D_B + w_\gamma.D_\gamma + w_n.D_n + w_p.D_p \qquad (5)$$

where

D_{bw}, D_B, D_γ, D_n, D_p: Weighted biological Effective Dose, Boron Dose, Gamma Dose, Thermal Neutron Dose and Proton Dose (Gy), respectively.

w_c, w_γ, w_n, w_p: weighting factor according to each type of radiation. The weighting factor w_c (c for combined) combines the effects of heterogeneous microdistribution of the boronated compound as well as the RBE arising from the high LET and Li particles.

ϕ_{th}: Thermal neutron flux (n/cm^2/s).

σ_N , σ_H and $\sigma_a^{B-10}(E_0)$: Thermal neutron cross-section for nitrogen, hydrogen and boron which are 1.7, 0.33 and 3839 barns, respectively.

w_{B-10}: Weight percent of ^{10}B (15ppm).

W_{B-10}: Atomic weight of ^{10}B (10.01293g/mol).

N_A : Avogadro number (6.02×10^{23} particles/mol).

Q : The energy imparted to the alpha and lithium ions (2.31MeV).

N_N and N_H : Number of nitrogen and hydrogen atom in 1kg of tissue, respectively.

E_n: Fast neutron energy (MeV).

In this calculation, the weight percent of boron is assumed to be 15ppm which is the average boron content in blood. The calculation is given in Table II and Figure 5. As we can confirm from the Figure, total dose is maximum at the surface of phantom and rapidly reduces with an increasing in the depth in phantom. Boron dose was extremely low because the thermal neutron flux is insufficient. The dose distribution in phantom is utilized in irradiation planning, such as determining the optimal size of the collimator and the optimal position for the patient’s head. A research to determine the 3D dose distribution in phantom in more detail and the effect of collimator shape and length will be carried out in the near future.

Table II. Total biological weighted effectiveness dose and its component versus depth in phantom.

No.	Depth in phantom (cm)	D_γ (Gy)	D_B (Gy)	Fast neutron dose (Gy)	Thermal neutron dose (Gy)	Total biological weighted effective dose (Gy)
1	1	1.18E-05	1.26E-07	7.27E-05	4.27E-06	2.58E-04
2	2	6.93E-06	7.38E-08	4.26E-05	2.50E-06	1.51E-04
3	4	2.21E-06	2.35E-08	1.36E-05	7.99E-07	4.83E-05

4	6	7.42E-07	7.91E-09	4.57E-06	2.68E-07	1.62E-05
5	8	3.07E-07	3.27E-09	1.89E-06	1.11E-07	6.71E-06
6	10	1.26E-07	1.34E-09	7.73E-07	4.53E-08	2.75E-06
7	12	4.51E-08	4.81E-10	2.78E-07	1.63E-08	9.87E-07
8	14	2.80E-08	2.98E-10	1.72E-07	1.01E-08	6.12E-07
9	16	1.14E-08	1.22E-10	7.04E-08	4.13E-09	2.50E-07

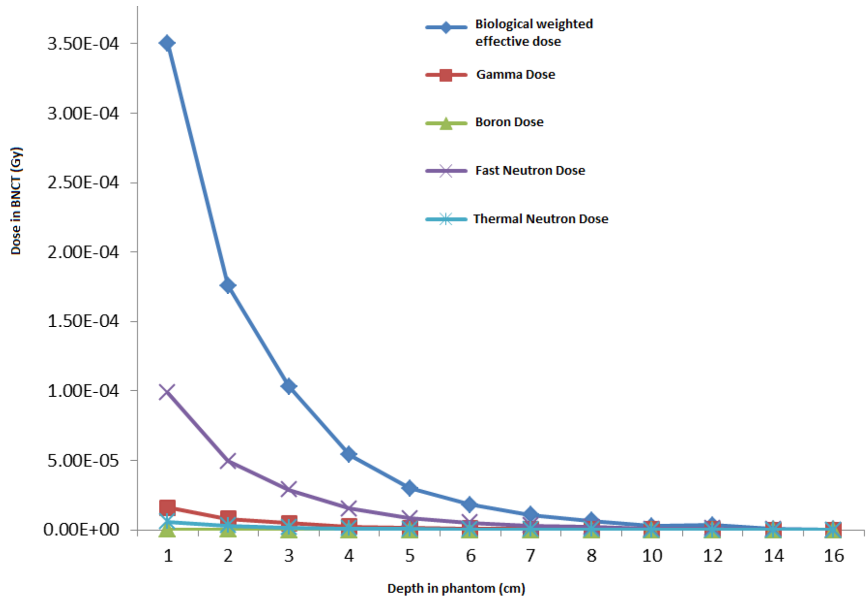


Fig. 5. Total biological weighted effective dose in BNCT and its component versus depth in phantom.

III. CONCLUSIONS

The aim of this research is to determine the radiation dose in the BNCT, using MCNP5 program for simulation and activation method for experimental measurement. The main results can be stated as follows:

- The thermal neutron flux distribution along the beam port's axis of Channel 2 of Dalat nuclear reactor was determined vial both experiment and simulation method.

- Radiation dose calculation for the BNCT using MCNP5 program was performed. The thermal neutron dose, gamma dose were much lower than the fast neutron dose. The radiation dose increased significantly once the phantom was closed to the beam port.

- The focal point of the neutron beam had significant effect on the dose distribution within the phantom.

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