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Monte Carlo Simulation Study for Boron Neutron Capture Therapy

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1. Introduction

Accelerator based boron neutron capture therapy system is compact, safe, and useful comparing with nuclear power plant based boron neutron capture therapy. Monte Carlo simulation was conducted about three parts in this presentation such as neutron flux after moderator assembly, stopping power of epithermal neutron and radiation shielding at accident situation.

2. Monte Carlo simulation about BNCT(Boron Neutron Capture Therapy)

2.1 Neutron flux and spectrum after moderator

If 8MeVproton beam interact with beryllium target, the neutron was generated. Figure. 1 shows that neutron flux was saturated over thickness of 500 um.

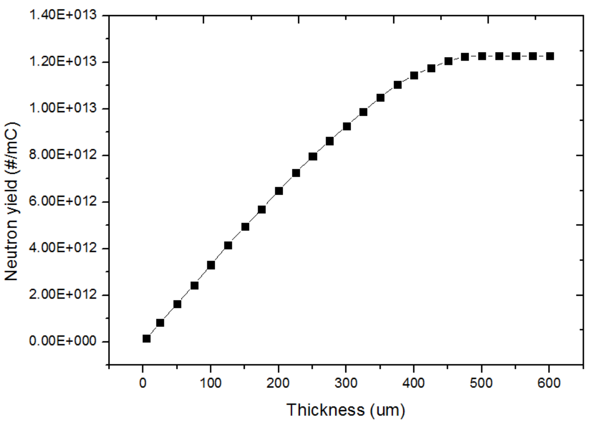
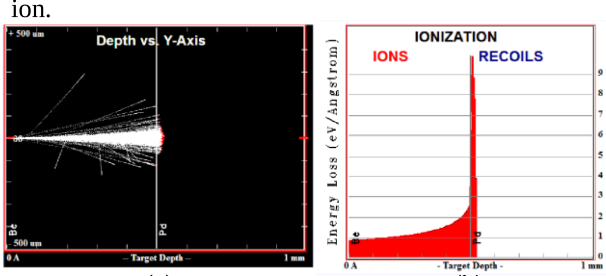


Figure. 1: MCNPX simulation of neutron yield as Be target

The TRIM simulation code was also used for calculation of beryllium thickness. Figure. 2 shows stopping range and particle trace by charged hydrogen ion.



(a) (b)

Figure. 2: (a) Proton particle track, (b) Energy loss in target by TRIM code

The generated neutron include lots of fast neutron. The fast neutron need moderator. Because the epithermal neutron need for the depth therapy by neutron with 1E+09 $\#/cm^2 \cdot sec$ on epithermal range. So, the material of moderator used nickel and AlF3 for generating the epithermal neutron. The Figure. 3 shows neutron energy spectrum after moderator.

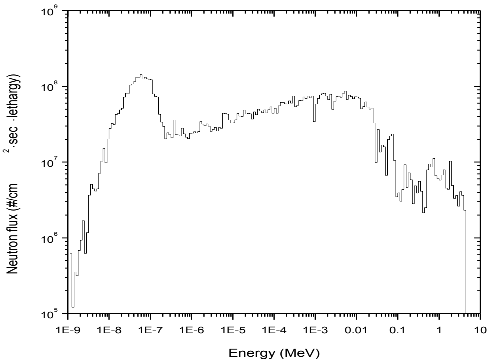


Figure. 3: Neutron distribution after moderator

2.2 Stopping power of epithermal neutron in water

The neutron beam has broad energy spectrum. But >50keV neutron need for the depth therapy. So, the stopping power was calculated at neutron of 50keV for the epithermal neutron by MCNP simulation. Figure. 4 shows schematic of MCNP simulation. And Figure. 5 shows neutron population and difference of population each cell.

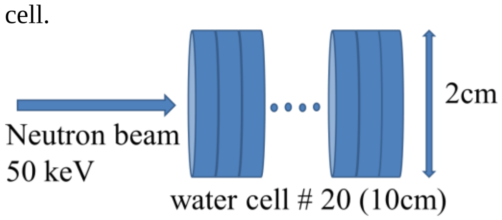


Figure. 4: The schematic of MCNP simulation

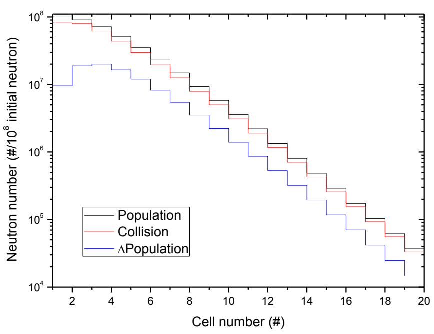


Figure. 5: Neutron population and differential of population in each cell

And simulation of the neutron capture in water cell was conducted for capture ratio in water cell. The capture number indicated on MCNP output. But, the number of nuclear reaction in cell need more information. So, capture number of MCNP output compared with tally multiplier collaboration with MT card for variable nuclear reaction. The figure. 6 shows the comparison MCNP output with FM and MT card.

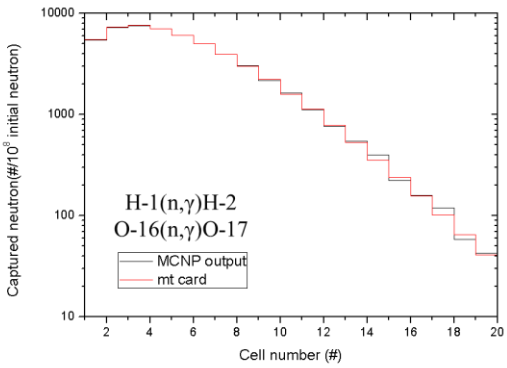
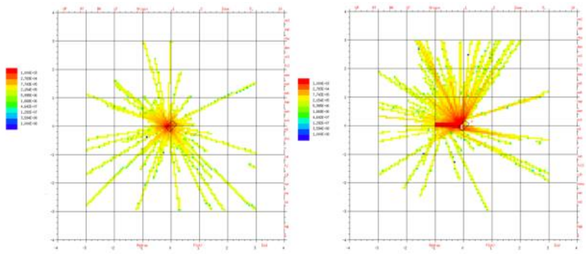


Figure. 6: Comparison MCNP output with MT card about neutron capture

2.3 Radiation shielding for neutron and photon source

In accident situation, the proton beam could not bend in magnet. So, proton beam interact with stainless steel of copper materials. The irradiated stainless steel and copper become the neutron and photon radiation source. The dose need for calculation by neutron and photon radiation. Figure. 7 shows neutron and proton mesh tally around target material.



(a) (b)

Figure. 7: Proton beam interact with SS304 (bending magnet) (a) neutron mesh tally, (b) proton mesh tally

And the Figure. 8 shows that dose rate of surface as concrete thickness. If the thickness of concrete over 200 cm, that is enough to normal treatment area.

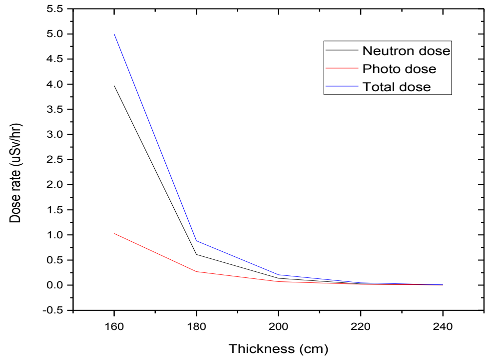


Figure. 8: Neutron, photon, and total dose rate at concrete surface

3. Simulation results

The materials of main moderator used nickel and AlF3. In Figure. 3, the flux peak of neutron indicated thermal neutron. So, geometry and materials types optimize to need for increasing epithermal neutron flux.

The neutron has not charge. So, neutron also has not bragg-peak in matter. The comparison results of neutron capture in MCNP output is similar with MT card.

This simulation study is base results for the Boron Neutron Capture Therapy. So, the results of simulation need enhancement for optimized epithermal neutron flux, depth dose of neutron, and radiation shielding.

Acknowledgment

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