DESIGN & SIMULATION OF A BNCT FACILITY AT MALAYSIAN NUCLEAR AGENCY WITH MCNP CODE

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ABSTRACT

The objective of this research is to optimally design an arrangement for BNCT facility at a beam line through the thermal column of Malaysian TRIGA MARK II Reactor. There is no similar research had been done at this thermal column and all TRIGA reactors have different characteristics. The characteristics of the neutron beam needed are thermal neutron with a flux of 109 cm-2s-1. Collimator, moderator and shielding components (room) with reactor core and thermal column will be simulated with Monte Carlo N-Particle Transport Code version 5 (MCNP5). Fluxes of neutron and photon will be calculated together with their energies with the same software.

ABSTRAK

Objektif penyelidikan ini adalah untuk merangka secara optimum susunan untuk kemudahan BNCT pada garis rasuk melalui ruang termal Reaktor TRIGA MARK II Malaysia. Tiada penyelidikan yang serupa telah dilakukan di litar terma ini dan semua reaktor TRIGA mempunyai ciri-ciri yang berbeza. Ciri-ciri rasuk neutron yang diperlukan adalah neutron haba dengan fluks 109 cm-2s-1. Collimator, penyederhana dan komponen perisai (bilik) dengan teras reaktor dan ruang haba akan disimulasikan dengan Kod Pengangkutan Monte Carlo N-Particle 5 (MCNP5). Fluks neutron dan foton akan dikira bersama-sama dengan tenaga mereka dengan perisian yang sama.

Keywords: boron, neutron, thermal column, reactor, collimator, moderator, shielding, photon

INTRODUCTION

An alternative cancer treatment that can treat cancer using neutron beams is called Boron Neutron Capture Therapy (BNCT). It is a very promising cancer treatment using the neutron radiation which can be obtained from either a low-flux nuclear research reactor or neutron generator and it is the interest of this study. BNCT exploits the selective deposition in tumor cells of boron carriers, boronophenylalanine (BPA) and sulfhydryl borane (BSH), enriched with 10B isotope, and the high thermal neutron capture cross-section of 10B. When a high boron concentration ratio between tumor and healthy tissue is reached, the patient is irradiated with low energy neutrons either thermal neutron or epithermal neutron (Durisi, 2015). When the tumor is irradiated with thermal neutrons a capture nuclear reaction is induced in 10B converting it to 11B, which decays by emission of an alpha particle (Faião-Flores et al., 2010).

BNCT is a binary radiation therapeutic modality for cancer treatment. As a binary treatment modality, BNCT is based on the reaction between the non-radioactive isotope 10B and thermal neutrons. Neutron is a particle contained and formed the nucleus and is neutral because it does not have electrical charge. Neutrons, especially thermal neutrons can be absorbed by atomic nuclei that they collide with, creating a heavier isotope of the chemical element as a result. In BNCT, 10B will capture thermal neutron and became unstable. It will then change to 7Li isotope after emitting α and γ each with respectively linear energy transfer (LET).

BNCT is done by firstly, a stable isotope of boron-10 (10B) is administered to the patient via a carrier drug and then the patient is irradiated with a neutron beam. 10B will then undergo the capture reaction 10B(n, α)7Li where 10B capture cross-section for thermal neutrons is 3840 barn (Valda et al., 2005). This is why thermal neutron is used in BNCT. There are many studies on BNCT using research reactors. Long term goal for this research is to develop a cancer treatment facility which is safe and practical by using neutron emitted by a low flux research reactor. Firstly, it needs to establish a suitable flux of neutron beams. For TRIGA types research reactors, thermal column is mainly design to produce thermal/epithermal neutrons which can be utilized for BNCT. It is, therefore, the thermal column of Malaysian TRIGA MARK II Reactor was used to produce thermal neutron source for this research. In order to build a BNCT facility outside the reactor, neutron collimator, neutron moderator and shielding for neutron and gamma-ray were required to ensure the safety and practicality of the procedure.

Collimator is needed to collimate neutron beam from thermal column and sending them to the target area outside the reactor wall. Material used in collimator must not either absorb or slowing down the neutron. Clark et al. (2009) had defined collimator as any device for producing a parallel beam of radiation. Neutron moderator is needed to reduce the velocity of fast and ephitermal neutrons which means reducing their energy to thermal energy producing more thermal neutron. Clark et al. (2009) had defined moderator as a substance that slows down free neutrons in a nuclear reactor, making them more likely to cause fissions of atoms of uranium-235 and less likely to be absorbed by atoms of uranium-238. Moderators are light elements, such as deuterium (in heavy water), graphite, and beryllium, to which neutron can impart some of their kinetic energy on collision without being captured. Neutrons that have had their energies reduced in this way are said to have been thermalized or to have become thermal neutrons. Neutron shielding is needed to avoid unwanted exposure of patient and radiation worker to the neutron. It is the same for gamma-ray. Clark et al. (2009) had defined shielding as a barrier used to surround a source of harmful or unwanted radiations.

MATERIALS AND METHOD

The one and only Malaysian nuclear reactor is situated in Bangi, Selangor State of Malaysia. The reactor type is TRIGA MARK II. The maximum power is 1 MW. This reactor is mainly used for research. It also serves for some industrial applications.

Simulation

The collimator of TRIGA MARK II reactor core and thermal column and shielding were simulated with Monte Carlo N-Particle Transport Code, Version 5 (MCNP5). After that, collimator and neutron and gamma-ray shielding (room) were added in the simulation. Fluxes of neutron and photon were then calculated together with their energies. MCNP5 code was used because the sources, photons or particles, materials for shielding and tally (data calculation) to be used can be selected. In addition, the running time can be made shorter by using collections of variance reduction techniques under Truncation Methods and Population Control Methods. The variance reduction techniques that can be used are energy cutoff, time cutoff, geometry splitting with Russian roulette, energy splitting/Russian roulette, time splitting/Russian roulette, weight cutoff/Russian roulette, weight window, exponential transformation, implicit absorption, forced collisions, source variable biasing, point and ring detectors, DXTRAN, and correlated sampling. Variance reduction techniques used in this simulation were energy cutoff, geometry splitting with Russian roulette, and weight cutoff/Russian roulette. Besides that, an extensive collection of cross-section data allowed the use of all intended materials. MCNP5 code also is capable in solving a complicated three-dimensional and time-dependent problem (X-5 Monte Carlo Team, 2003).

MCNP5 code

In this research, simulation is crucially important as a guideline for good geometry, best material, proper design and especially safety precautions. Simulation gives expected result base on geometry, design and material used. Corrections can be made until good result is achieved. General-purpose MCNP code can be used for neutron, photon, electron, or coupled neutron/photon/electron transport, including the capability to calculate eigen values for critical systems and this code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first and second-degree surfaces and fourth-degree elliptical tori (X-5 Monte Carlo Team, 2003).

As a general-purpose, continuous energy, generalized geometry, time dependent, coupled neutron/photon/electron Monte Carlo transport code, it can be used in several transport modes which is neutron only, photon only, electron only, combined neutron/photon transport where the photons are produced by neutron interactions, neutron/photon/electron, photon/electron, or electron/photon. The neutron energy regime is from 10-11 MeV to 20 MeV for all isotopes and up to 150 MeV for some isotopes, the photon energy regime is from 1 keV to 100 GeV, and the electron energy regime is from 1 keV to 1 GeV (X-5 Monte Carlo Team, 2003).

Fluxes of Neutron and Gamma-ray Calculation

Neutron fluxes calculation will be done with MCNP5 code. The detector results are generally reliable if their associated relative errors are below 5%. The tally fluctuation charts at the end of the output file are base their results on the information from one specified bin of every tally. This bin also is used for the weight window generator and is subject to ten statistical checks for tally convergence, including calculation of the VOV. By using the DBCN card the VOV can be printed for all bins in a tally. Only when it passes all ten statistical checks a tally is considered to be converged with high confidence (Pelowitz, 2008).

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