

SIMULATION OF NEUTRON FLUX IN SILICON, CADMIUM AND PLUMBUM USING
MONTE CARLO METHOD

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ABSTRACT

The purpose of this study is to investigate the neutron energy flux from reactor after attenuated by silicon, cadmium and plumbum at the end of the beam port. Neutrons can be categorized into thermal neutron, intermediate and fast neutron according to their energies. Fast neutrons in reactor needs to be thermalized before it can be utilized for applications such as in medicine, industrial and research purposes. The fast neutron will lost its energy into thermal neutrons due to scattering and absorption process with nucleuses. The neutron probability of interaction with nucleus depends on the microscopic cross-section, which is different for each material. All the cross-section data for every element have been compiled in ENDF (Evaluated Nuclear Data File) format and internationally recognized. In this study, the neutron transport was simulated using Monte Carlo N-Particle Transport Code, Version 5 (MCNP5). The thickness of the materials used in this research is in the range of 1 cm to 10 cm. The result shows that the neutron was reduced significantly by silicon then follows by plumbum and cadmium. The thermal neutron flux was the lowest in cadmium because it have high thermal neutron cross-section.

ABSTRAK

Tujuan kajian ini adalah untuk melihat perubahan fluks dari reaktor selepas ia melalui silikon, kadmium dan plumbum di hujung laluan keluar neutron. Neutron boleh dikategorikan kepada neutron terma, neutron pertengahan dan neutron laju mengikut tenaga masing-masing. Neutron di dalam reaktor perlu disederhanakan kepada neutron terma sebelum ia boleh diaplikasikan kepada bidang perubatan, industri dan untuk tujuan penyelidikan. Neutron laju akan kehilangan tenaga dan menjadi neutron terma melalui proses serakan dan serapan dengan nukleus. Kebarangkalian neutron berinteraksi bergantung kepada keratan rentas mikro yang berbeza bagi setiap bahan. Data keratan rentas setiap bahan telah dikumpulkan di dalam format fail ENDF (Fail Evaluasi Data Nuklear) yang diakui di peringkat antarabangsa. Di dalam kajian ini pergerakan neutron di simulasikan dengan menggunakan Monte Carlo N-Particle Transport Code, Version 5 (MCNP5). Ketebalan setiap bahan yang digunakan di dalam kajian ini adalah diantara 1 cm ke 10 cm. Keputusan menunjukkan fluks neutron telah dikurangkan apabila melalui silikon dan plumbum. Fluks neutron terma adalah yang paling sedikit apabila melalui kadmium kerana keratan rentas serapan neutron terma adalah tinggi bagi kadmium.

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LIST OF SYMBOLS

α	-	Alpha particle
β	-	Beta particle
μ	-	Cosine angle
θ	-	Angle of collision
σ	-	Microscopic cross-section
ϕ	-	Neutron flux
χ	-	Coefficient
ρ	-	Density
ρ_0	-	Distance between two fragments
Σ	-	Microscopic cross-section
Ω	-	Angular cross-section
ν	-	Number of neutron generated in the fission
γ	-	Gamma ray

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CHAPTER 1

INTRODUCTION

1.1 Research Background

Neutron is produced from by the interaction of other particles with the nucleus. The particles can be alpha particles, gamma-rays, protons or even by the neutron itself. Neutron in the nucleus are binds together by nuclear force which acting in the distance of 1.5 fm. To break the force, the incident particle must have enough energy. But for some light elements i.e. beryllium, boron and lithium, the binding energy is low and the neutron can be ejected by the above mention particle in relatively low energies. Neutrons produced in this reaction will have a monoenergetic spectrum and the value depends on the type of incident particle and the type of target materials.

For some heavy element i.e. uranium and plutonium, if a neutron absorbed by this materials, they can undergo fission process and release a neutron in a chain process with energies in the range between 1 MeV to 12 MeV. For the purpose of application in the industrial, medical and other fields, the number of neutron flux or the energetic neutron needs to be decreased. Neutron with energy more than 1 MeV needs to be slow down to 0.025 eV before it can be used. Fission neutrons are widely used in the nuclear reactor to produce electricity around the world.

In a nuclear reactor several types of potentially dangerous radiation produced along with the fission process. The fission process emit α particles, β particles and recoiling fission fragments which carried electric charged. This causes the particles to be stopped immediately within the reactor or when travelled in a short distance inside the shielding in the form of heats. Neutrons, neutrinos and γ rays emitted from the reactor are neutral particles. Neutrinos can escape from the reactor without any interaction but neutrons and γ rays are the major concern in the principle of reactor nuclear shielding because they are neutral can penetrate easily in materials and cause damages. Neutrons produced in a reactor can be categorized into four groups:

1. Prompt fission neutrons.
2. Delayed fission neutrons.
3. Activation neutrons.
4. Photoneutrons.

Prompt fission neutrons emitted almost simultaneously in the fission event and delayed neutrons emitted after a very short time ($\ll 1 \mu\text{s}$) which resulted from the decay product of highly excited fission fragments. It is only 1% from the total emission of neutron with energy in the order of 0.5 MeV or less. Both prompt and delay neutron from thermal-fission can be represented by a Watt spectrum:

$$N(E) = 0.453e^{-E/0.965} \sinh(2.29E)^{1/2} \quad (1.1)$$

$N(E)$ is the fraction of neutrons per unit energy range emitted per fission, and E is the neutron energy in MeV. This formula originally the work done by Watt in the range of 0.075 to 17 MeV but re-examined by Cranberg et al. over the range 0.18-12.0 MeV as shown in the equation (1.2) and also known as the Maxwellian distribution [2].

$$N(E) = CE^{1/2} \exp^{E/1.2895} \quad (1.2)$$

In this spectrum, the neutrons are assumed to be emitted from a single fragment isotropically with a Maxwellian velocity distribution in the center-of-mass system. From this distribution, the average energy released is about 2 MeV and the neutron intensity maximum at neutron energy of 0.73 MeV. The energy spectrum could reach maximum energy of 10 MeV and above. Neutrons can be categorized into slow neutron, epithermal neutron and fast neutron according to their energies. In a reactor, epithermal and fast neutron needs to be moderated into slow neutron before ^{235}U can capture the neutron and sustain the fission chain. Some of the neutrons will be directed into beam ports to be utilized for applications as mentioned above. This neutron flux also needs to be moderated to suitable energy depend on the application.

The principle of shielding is based on the collision and absorption process between neutrons and the material. The neutrons will slow down and if absorption occurred, the number of neutrons will be reduced. The probability of interaction between neutron and matter can be measured by its area of interaction or called as neutron cross-section. Neutron cross section values are different for each interaction i.e. elastic scattering, inelastic scattering or absorption. When a neutron beam transport in a medium it can produce γ photons then photons liberate electrons, electrons give rise to X-rays that will liberate electrons, etc. All this process can be described by the Boltzmann radiation transport. This is an integro-differential equation that can be solved by several mathematical ways. Monte Carlo method is one of the ways to solve the transport equation stochastically using probability distribution.

Monte Carlo is a computer programming for solution to radiation transports equation. It is based on random sampling method to solve the mathematical diffusion theory and give the answers in mean of statistical distribution. The randomness of statistical probability of the result is where the Monte Carlo get its name, just like the games of chances in the world's famous casino. The Monte Carlo method was developed during the World War II at Los Alamos in which the nuclear weapon becomes crucial. It was developed by mathematician, John von Neumann together with Nicholas metropolis, Stan Frankel, Enrico Fermi, Stan Ulam and many others. This method can be used successfully only after the development of electronics computer at time around 1940s and 1950s.

1.2 Problem of Research

This research will take TRIGA reactor at Nuclear Malaysia Agency, Bangi, Selangor as a reference because this reactor produce neutrons with Maxwellian energy distribution. TRIGA is stand for Training, Research, Isotope Production and General Atomic. This reactor was started built in 1970s in preparation for Malaysia developing a nuclear power for electricity [3]. However, oil was discovered in the east coast of Peninsular Malaysia in 1974 and after that Nuclear Malaysia only organized the reactor for non-power research mainly in nuclear physics, radiation chemistry, biology, isotope production, shielding studies and training.

The thermal neutron flux produced in the TRIGA reactor core at the maximum is 1×10^{12} n/cm²/s. Then it is reduced to about 10^8 n/cm²/s when it enters the beam port. It is important to have neutron flux at a tolerable level for operations and applications as

mention before, or for biological shielding in the working areas. The interaction between neutrons and materials can reduce the neutron energy and its flux and the effects are different for each material. To reduce the neutron energy and also the neutron flux, a certain material is placed at the end of the beam port.

1.3 Research Objective

The objectives are:

1. To investigate the effect of cross-section and thickness for silicon, cadmium and plumbum in reducing the neutron flux from TRIGA reactor.
2. To find the optimal thickness in reducing the neutron flux from the ranges of 1 cm to 10 cm for cadmium and plumbum.

1.4 Scope of Studies

The geometry in this research used the TRIGA MARK II Reactor as a reference. In this research three materials, which are plumbum, cadmium and silicon were chose as to decrease the neutron energy and its flux in the beam port. The length of the materials was varied from 1 cm to 10 cm with 10 cm in each interval. The neutron flux was simulated using Monte Carlo N-Particle Code Version 5 developed by Los Alamos National Laboratory.

The neutron flux from fission process in the core of the TRIGA reactor was directed into the beam port and neutron's flux was recorded at the end of the beam port after moving through the above mentioned materials. In this research, the energy tally for neutron is from 0.02 eV to 8 MeV, whereas the energy spectrum from the core used the Maxwellian distribution is in the range 0.002 eV to 12 MeV. The neutron flux tracking was simulated and recorded by MCNP using tally 2 ($f2:n$).

One of the objectives of this study is to find the neutron flux after passing through the silicon, cadmium and plumbum. These materials were placed at the end of the beam port and the fluxes of the neutron are recorded. All the flux from each material was compared to each other.