การวิเคราะห์ฟลักซ์นิวตรอนจากเครื่องปฏิกรณ์ด้วยวิธีการมอนติคาร์โล

รพพน พิชา

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บทคัดย่อ

แบบจำลองการฉายรังสีสสารด้วยรังสีนิวตรอนจากเครื่องปฏิกรณ์ได้ถูกสร้างและทดสอบในโปรแกรม MCNP เพื่อใช้ในการประมาณปริมาณฟลักซ์ที่สสารที่มารับการอาบรังสีจะได้รับ ในการทดลองนี้ วัตถุสามชนิด คืออลูมินัม แคดเมียม และแทนทาลัม ได้ถูกใช้เป็นสิ่งกั้นระหว่างแกนเครื่องปฏิกรณ์และตัวสสาร ผลที่ได้จากการคำนวณได้แสดงว่า แคดเมียมสามารถลดปริมาณนิวตรอนได้สูงกว่าแทนทาลัม โดยเฉพาะในช่วงพลังงานเทอร์มัล

คำสำคัญ: นิวตรอน ฉายรังสี เครื่องปฏิกรณ์ วิธีการมอนติคาร์โล

Analysis of Neutron Flux Using Monte Carlo Methods

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Abstract

The energy profile of neutrons from a fission reactor core and a neutron irradiation setup are simulated. The neutron doses deposited inside casings of aluminum, cadmium, and tantalum are studied via MCNP simulations to estimate the doses received by materials with different types of shielding. It is found that the difference in dose reduction between cadmium and tantalum is most pronounced at the thermal energy region.

Keywords: neutrons, reactor, irradiation, Monte Carlo

Introduction

Nuclear fission processes which occur inside the core of a nuclear reactor create neutrons with a wide range of energy. These neutrons can be used in research as well as many radiation utilization techniques such as isotope creation, neutron activation analysis (NAA), thin-film etching, and material modification. Gemstone enhancement is also one of the main applications of neutron rays. Thailand's Office of Atoms for Peace (OAP) has a facility to perform irradiation using the neutrons from its 2-MW research reactor (TRR-1/M-1). The required length of irradiation time depends on source strength and distance between the gemstone and the neutron source.

In this study we looked into the neutron doses deposited inside a material enclosed by different types of metal casings. Neutron flux from a fission source was simulated using MCNP (Monte Carlo N-Particle) software package. The computer program uses Monte Carlo methods to simulate interaction of radiation with matter. MCNP has been extensively tested against experimental results, and the validation has been found to be satisfactory [1, 2]. Therefore by running simulations we will be able to design different kinds of actual irradiation experiments efficiently.

MCNP was used to calculate neutron fluxes at various distances and inside different kinds of containers. Geometry is based on an actual experimental setup. A cylinder rod simulates a fuel rod inside the reactor. A metal tube is constructed with dimensions of an actual gemstone container. Light water is used as the medium between the neutrons and the tally volumes.

The collected results (tallies) were binned by energy. Neutrons can be classified by their energies into various groups: cold, thermal, epithermal, slow (resonance), and fast. The designation criteria can be quite arbitrary. Typically, cold neutrons have energy below 0.005 eV, thermal neutron have energy about 0.025 eV (equivalent to thermal energy of gas molecules at room temperature), epithermal between 0.5 eV-1 keV, slow neutrons 1-100 keV, and fast neutrons have energies over 100 keV. In this study we call neutrons with energy $\leq$ 1 eV thermal, those with 1 eV $< E \leq$ 1 keV epithermal, and refer to anything above as fast neutrons.

Methods

Two types of studies were performed. First we looked at the effect of distance on the neutron flux inside an Al tube. Then we investigated into how a secondary material affects the flux.

MCNP geometry is constructed via cells and surfaces. Several different cells were needed: one cell for water, one for the source, one for the water inside the metal tube. In the distance study we needed one cell for the cylindrical shell, while the material study needed two cells for 2 coaxial shells. The "material test" setup is shown in Figure 1. The "distance test" used a similar setup, but without the inner metal shell.

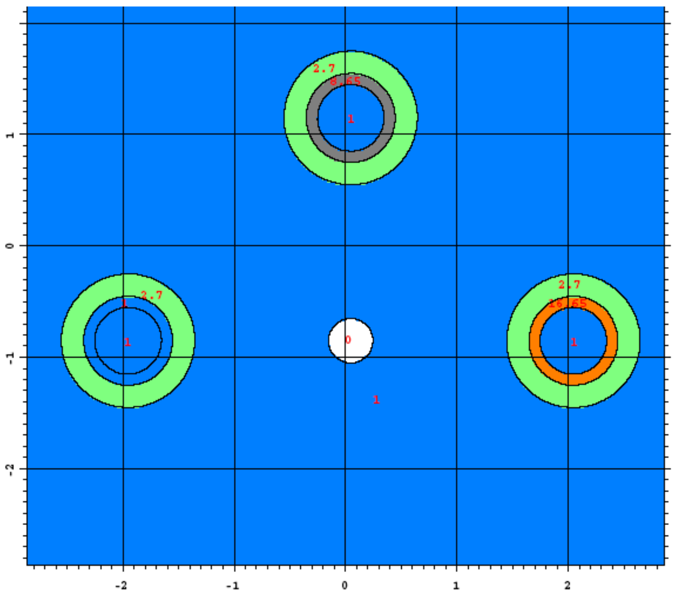

Figure 1: Top-view geometry of the setup. We have the neutron source in the middle and three cylindrical containers at equal distances from the source.

There are three types of containers placed at equal distances from the neutron source. Each container has an aluminum shell as the first layer. The aluminum shell is 2 mm thick, with a 5-mm radius. The fuel rod's radius is 2 mm. Both the Al cylinder and the fuel rod are 2-cm high. One container has a 1-mm cadmium shell inside; one has a 1-mm tantalum shell; and one does not have any extra shell–this is our reference container. Water is used everywhere else.

Aluminum of different densities were tested. The density of aluminum in the distance study was set at 4.0 g/cm^3 while in the second study we used 2.7 g/cm^3 . The density effect on the flux is negligible.

Neutron absorption and total cross sections of various materials used in this analysis are shown in Figure 2. The direct comparisons of the cross sections are shown in Figure 3. From this result, we can see that cadmium appears to be a very good thermal neutron absorber.

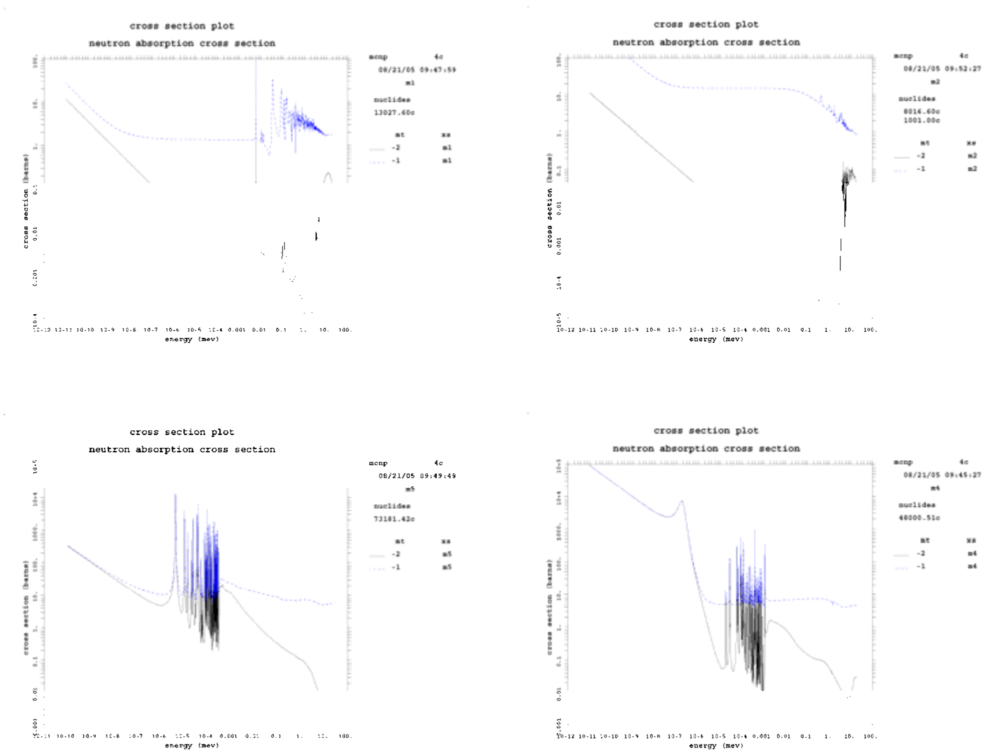

Figure 2: Absorption and total cross sections of neutrons inside materials. Clockwise from upper left: aluminum, water, cadmium, and tantalum.

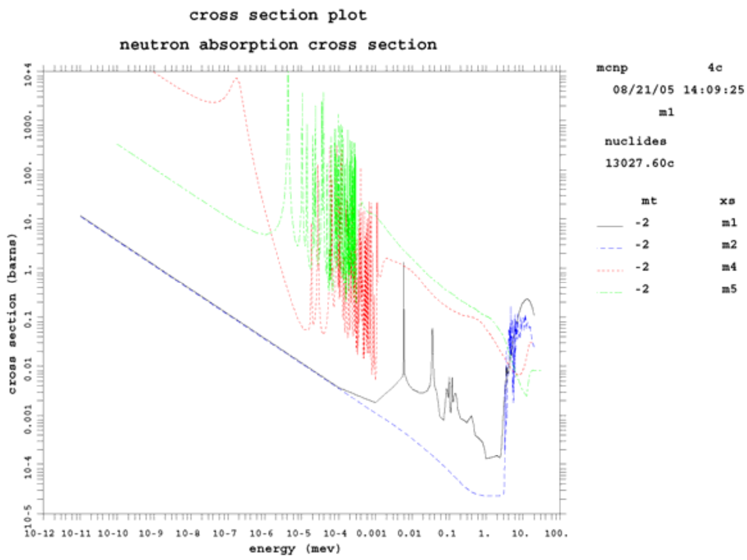

Figure 3: Comparisons of neutron absorption cross sections in different materials.
m1 is aluminum (2.7 g/cm^3); m2is water (1.0 g/cm^3);
m4 is cadmium (8.65 g/cm^3); m5 is tantalum (16.65 g/cm^3).

The energy distribution of the source neutrons used in this study is the Watt fission spectrum:

$$p(E) = C \cdot \exp(-E/a) \cdot \sinh(\sqrt{bE})$$

with parameters a = 0.988 and b = 2.249, which represents the case of ^{235}U fissions by thermal neutrons. E is in MeV. Figure 4 shows the distribution shape. (Normalization factor C is set to 1.)

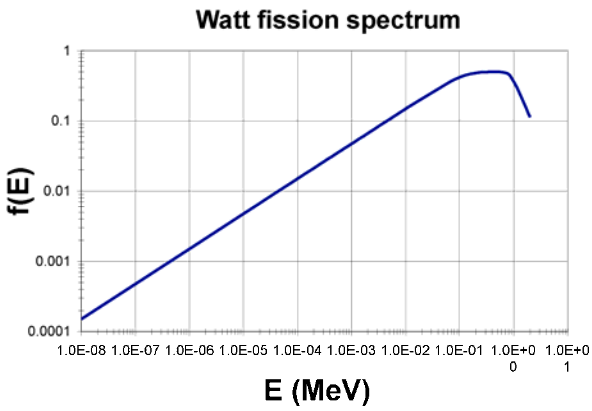

Figure 4: Watt fission energy distribution.

Results and Discussion

Figure 5 shows the flux of neutrons inside the aluminum tube, averaged per neutron started.

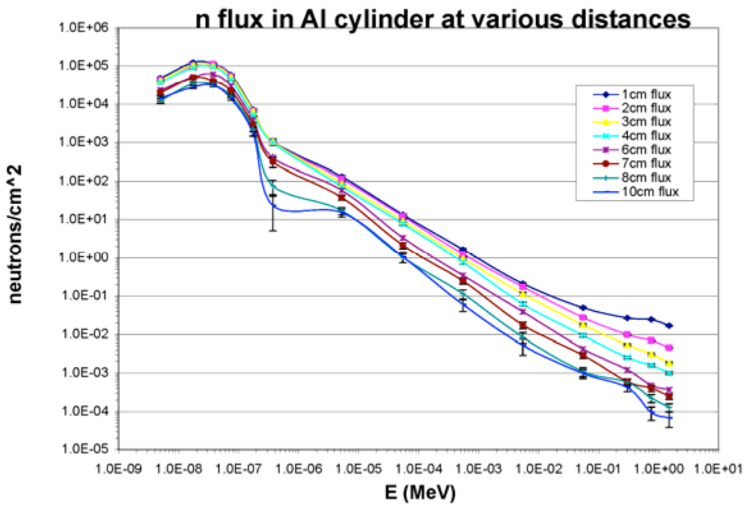

Figure 5: Neutron flux at various distances.

The number of neutrons is averaged per neutron started.

Figure 6 shows the neutron flux when we use a 2mm-thick Al tube, 2mm-thick Al plus 1mm-thick coaxial Ta tube, and 2mm-thick Al plus 1mm-thick coaxial Cd tube. It is clear that cadmium is an effective absorber of low-energy neutrons, as already suggested previously by the neutron absorption cross section plot. A 1-mm thick Cd shell is able to absorb over 90% of thermal neutrons that pass through the Al shell.

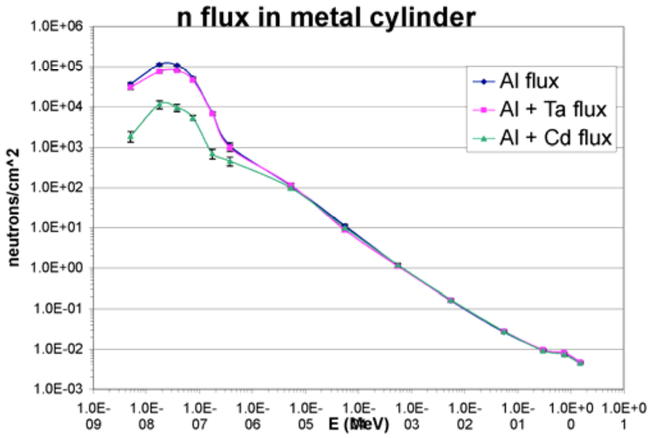

Figure 6: Neutron flux inside different kinds of metal tubes.

The number of neutrons is averaged per neutron started.

By using flux-to-dose conversion in MCNP, we built a dose profile in each of the different cases. In Figure 7 the neutron absorbed doses are shown. The ratio between the Cd+Al and the Ta+Al cases is plotted in Figure 8. As in the neutron flux plot, the difference in neutron absorption is most clearly observed at low energy region.

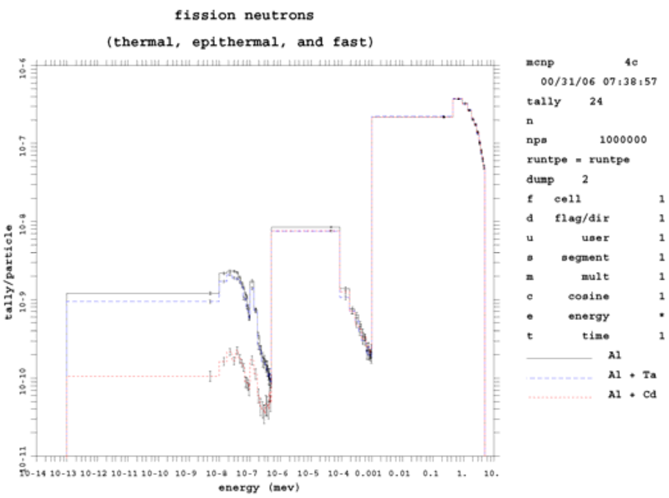

Figure 7: Equivalent doses inside different kinds of metal tubes. The unit of the equivalent dose is rem per hour per unit flux.

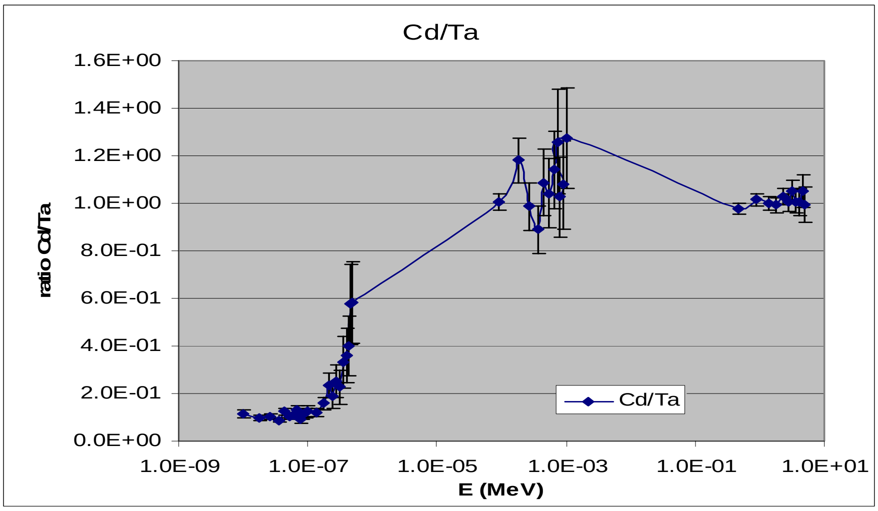

Figure 8: Ratio of neutron doses inside Cd/Al and Ta/Al containers.

Conclusion

Neutron fluxes at different distances were plotted. Effects from cadmium and tantalum shells were shown. Cadmium is better than tantalum at absorbing thermal neutrons. Tantalum is slightly better in the epithermal range. For higher energy (E>1 keV) neutrons, neither of them is an effective neutron absorber. Future studies involving more types of materials can be conducted to make a more precise quantitative comparison.

References

1. Forrest B. Brown, Russell D. Mosteller, and Avneet Sood, "Verification of MCNP5," Proc. M&C 2003: A Century in Review, A Century Anew, Gatlinburg, Tennessee (April 2003).
2. Russell D. Mosteller, "Validation Suites for MCNP™," Proc. 12th Biennial Topl. Mtg. Radiation Protection and Shielding Div., Santa Fe, New Mexico (April 2002).