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# Application of Monte Carlo Method to Design a Delayed Neutron Counting System

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## 1. Introduction

The quantitative determination of fissile materials in environmental samples is becoming more and more important because of the increasing demand for nuclear nonproliferation. A number of methods have been proposed for screening environmental samples to measure fissile material content. Among them, delayed neutron counting (DNC) that is a nondestructive neutron activation analysis (NAA) method without chemical preparation has numerous advantages over other screening techniques.[1] Fissile materials such as $^{239}$Pu and $^{235}$U can be made to undergo fission in the intense neutron field. Some of the fission products emit neutrons referred to as "delayed neutrons" because they are emitted after a brief decay period following irradiation. Counting these delayed neutrons provides a simple method for determining the total fissile content in the sample. In delayed neutron counting, the chemical bonding environment of a fissile atom has no effect on the measurement process. Therefore, NAA is virtually immune to the "matrix" effects that complicate other methods.[2]

The present study aims at design of a DNC system. In advance, neutron detector, gamma ray shielding material, and neutron thermalizing material should be selected. Next, investigation should be done to optimize the thickness of gamma ray shielding material and neutron thermalizing material using the MCNPX that is a well-known and widely-used Monte Carlo radiation transport code [3] to find the following

## 2. Methods and Results

### *2.1 Neutron detector*

Irradiated samples emit neutrons and gammas. So the selected neutron detector should have not only high neutron detection efficiency but also good gamma discrimination ability. Among the commonly used neutron detectors, gas detectors are suitable for serving both purposes.[4] So He-3 detector was selected as candidate neutron detector for the DNC system. He-3 detector usually has a high neutron detection efficiency due to the high thermal neutron cross-section of He-3 (n,p) reaction. In addition, the high Q value of this reaction assures He-3 detector to discriminate gammas effectively. Further more, photons have much less chance to interact with gas than with solid or liquid, so there would be only insignificant number of spurious pulses that are caused by direct interactions between photons and gas molecules in detector's sensitive volume.

### *2.2 Gamma ray shielding material and thickness*

In strong gamma fields, He-3 detector may lose neutron counting ability due to the effect of pulse pileup. As the increase of gamma dose rate, pulse pileup becomes more and more serious so that smaller photon pulses are summed up to form bigger ones in neutron detectors. Eventually these pulses may exceed the preamplifier's energy discriminator threshold and be counted as neutron events. It has been shown that He-3 detector fails to work properly if gamma dose rate is too high. Specifically, a conventional He-3 detector works well only at gamma exposure rate below 10 rad/h.[4] Therefore, gamma shielding is usually necessary in neutron measurement in strong gamma fields to achieve a better signal to noise ratio. So lead was used as gamma shield. To calculate the thickness of lead, the gamma ray source term due to neutron irradiation was computed using the MCNPX and the following assumptions: (a) An environmental sample is 13.6 g of soil including 1 mg of $^{235}$U in the rabbit, (b) Irradiation is done at HANARO with $2.383 \times 10^{13}$ n·cm$^{-2}$·s$^{-1}$ during 60 seconds.

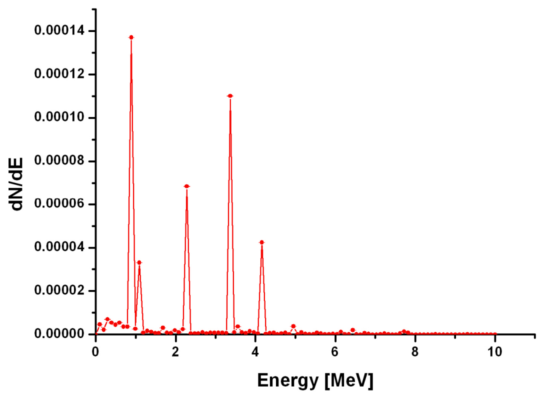


Fig. 1 Photon source spectrum computed with MCNPX.

Figure 1 is the computed gamma ray source spectrum. Investigation was done to optimize the thickness of lead using same code and computed gamma ray source. The average photon fluxes and exposures were determined for the various thickness of lead shield from 0 to 8 cm and the result is shown in figure 2. It was found that 5 cm lead is sufficient to reduce the gamma exposures inside the He-3 detector below the limits.

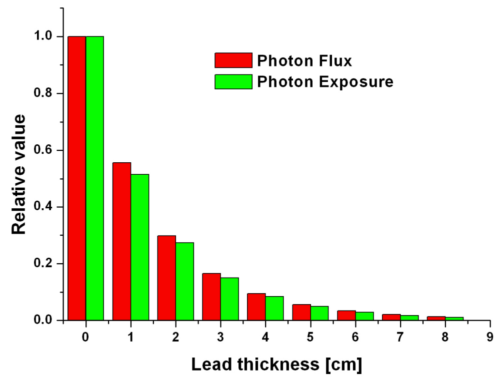


Fig. 2 Relative photon fluxes and exposures at detector position as a function of lead thickness.

### *2.3 Moderation material and thickness*

The DNC system should have moderator because energy of the delayed neutrons has range from thermal to 4 MeV. Various moderator options were considered based on moderation effectiveness, structural strength, manufacturability, and cost. The primary concern of the design was moderation effectiveness, thus initially polyethylene, water, concrete, and paraffin were investigated. Water and paraffin were eliminated to avoid possible maintenance concerns associated with a leakage and flammability respectively. Polyethylene and concrete were comparable moderator options. Both materials had similar moderating capabilities, were relatively inexpensive, were easily manufacturable, and were structurally feasible. One advantage for using polyethylene as opposed to concrete is its low density, approximately three times less, which would have delivered less weight on the building floor. However, a moderator composed of polyethylene would have required a structural support system. Therefore, meeting the initially described criteria to the best achievable level, polyethylene was chosen as the moderator material.

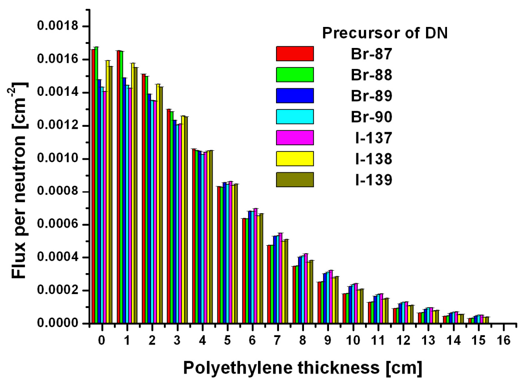


Fig. 3 Neutron fluxes at detector position as a function of polyethylene thickness.

Several MCNP simulations were executed using the delayed neutron spectrum of precursors in the ENDF/B-VI data base. These simulations were run changing thickness of polyethylene with lead shield. The outer side of the moderator was monitored with a volume detector same size as He-3 detector, which calculated the thermal neutron flux. These results are shown in figure 3. Based on figure 3, a moderator with 1 or 2 cm was approximately optimized.

Finally, The DNC system consists of 17 He-3 detectors embedded in a cylindrical configuration inside polyethylene. In the center of the system there is a through hole with diameter of 10 cm to install the sample flight tube. Next to the hole is a 5 cm thick lead for gamma shielding. The distance between the detector surface and the inner surface of the polyethylene is 1.3 cm. The outer diameter of moderator is 50 cm and its length is 50 cm to cover whole detector. The system is shielded against the neutron background by a 0.5 mm thick of cadmium sheets. The DNC system designed in the present study is shown in figure4.

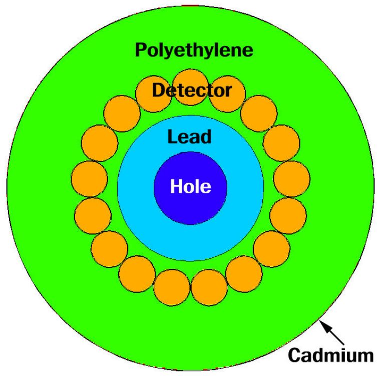


Fig. 4 Cross sectional view of the DNC system.

## 3. Conclusion

The delayed neutron counting system was designed to estimate of the fissile material with Monte Carlo method. The optimized DNC system consists of 17 He-3 detectors embedded in a cylindrical configuration inside polyethylene for neutron moderation. In the center of the system there is a through hole with diameter of 10 cm to install the sample flight tube. Next to the hole is a 5 cm thick lead for gamma shielding. The distance between the detector surface and the inner surface of the polyethylene is 1.3 cm. The outer diameter of moderator is 50 cm and its length is 40 cm to cover whole detector. The system is shielded against the neutron background by a 0.5 mm thick of cadmium sheets.

## REFERENCES

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