ANALYSIS OF A CADMIUM-PLASTIC SCINTILLATION CAPTURE-GATED NEUTRON DETECTOR

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ABSTRACT

Efficient, robust neutron detection systems are paramount in nuclear safeguards and nonproliferation applications. Nearly all commercially applied neutron counting systems employ ^{3}He gas for the detection of thermal neutrons. These systems provide high neutron detection efficiency while being almost completely insensitive to gamma rays. However, due to the large number of such systems deployed in the field, there is currently a shortage of ^{3}He gas. Therefore, novel neutron detection systems are needed. In order to be competitive with ^{3}He, these systems should preserve the incoming neutron energy information while maintaining high detection efficiency and allowing neutron/gamma ray discrimination. One class of detectors that has been recently explored is capture-gated scintillation detectors which consist of a standard scintillation detector that has been modified to include a neutron capturing isotope. Here, a novel detector is described consisting of slabs of commercial plastic scintillator separated by thin foils of cadmium metal. The combined signal generated by multiple (n, p) collisions in the plastic scintillator is a measure of the incident neutron kinetic energy if the neutron deposits all of its energy within the detector body. Low-energy neutrons that remain in the detector may be captured in the cadmium sheets (the thermal-neutron cross section is approximately 2,500 b for natural cadmium). A total of 9 MeV of gamma-ray energy is released upon capture and these gamma rays will provide a large scintillation pulse. Simulations of this detector will be performed using the MCNP-PoliMi code. A prototype of this detector has been constructed recently. In this paper, both simulated and measured results are presented.

Keywords: ^{3}He replacement; capture-gated scintillator; MCNP-PoliMi

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1. INTRODUCTION

Nearly all commercially applied neutron counting systems employ ^{3}He gas for the detection of thermal neutrons with high efficiency while being almost completely insensitive to gamma rays. However, due to the large number of such systems deployed in the field, there is currently a severe shortage of ^{3}He gas. Therefore, novel neutron detection systems are needed. In order to be competitive with ^{3}He, these systems should preserve the incoming neutron energy information while maintaining high detection efficiency and allowing neutron/gamma-ray discrimination. One class of detectors that has been recently explored is capture-gated scintillation detectors which consist of a standard scintillation detector that has been modified to include a neutron capturing isotope. Here, one such detector based on cadmium and plastic-scintillator will be characterized using both simulations and measurements for application in neutron spectroscopy for potential ^{3}He replacement.

2. CAPTURE-GATED SCINTILLATION DETECTORS

A capture-gated scintillation detector is a traditional scintillation organic detector (either liquid or plastic) that also contains a thermal-neutron absorbing isotope (for example ^{10}B, ^{6}Li, ^{nat}Gd, or ^{nat}Cd) [1]. Due to the large (approximately 50 atom-percent) hydrogen content, the scintillation material acts as an effective neutron moderator. Once the source neutrons have been sufficiently down-scattered they will, with relatively high probability, be absorbed in the capturing material. The moderation process in the scintillation material produces a light pulse, which, when analyzed in coincidence with a capture pulse, can be assumed to contain all of the incident neutron energy. This is because the neutron capture occurs with highest possibility once the neutron has been completely thermalized. Selective analysis of these pulses may allow capture-gated scintillation detectors to provide enhanced neutron spectroscopic capability compared to traditional scintillation detectors.

One such detector is composed of plastic-scintillator layers separated with thin layers of ^{nat}Cd metal (concept developed by J. B. Czirr at Brigham Young University). The ENDL92 total absorption cross section of ^{nat}Cd is shown in Figure 1. A total of 9 MeV of gamma-ray energy is released upon a single capture event and these gamma rays can provide a strong capture signal. The combined signal generated by multiple (n, p) collisions in the plastic scintillator is a measure of the incident neutron kinetic energy if the neutron deposits all of its energy within the detector body. Low-energy neutrons that remain in the detector may be captured in the cadmium sheets. The larger the detector body, the more gamma-ray energy is captured through numerous Compton-scatter events. A prototype of this detector has been constructed. It consists of 13 1-cm thick BC-408 plastic scintillator layers and 12 0.1-mm thick Cd layers. An illustration of this prototype is shown in Figure 2.

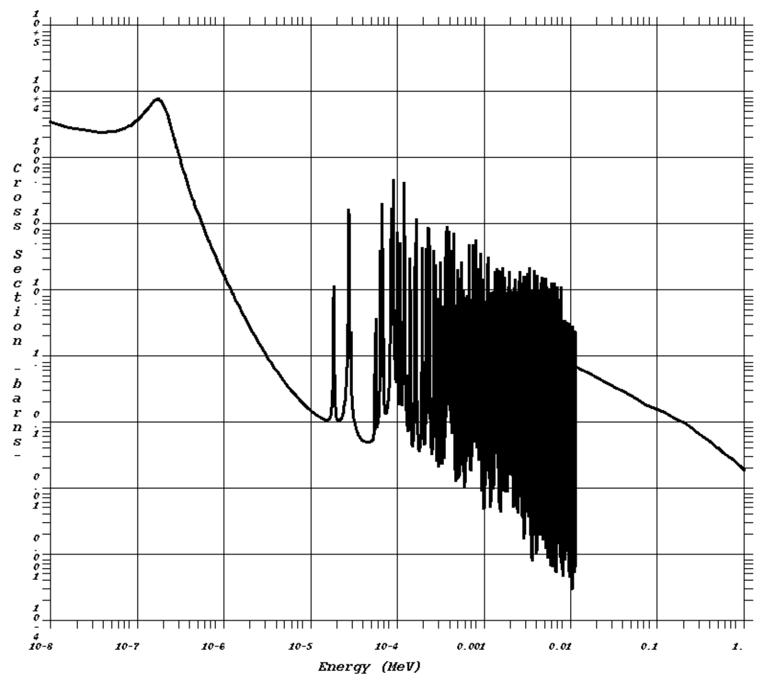

Fig. 1. ENDL92 ^{nat}Cd total absorption cross section.

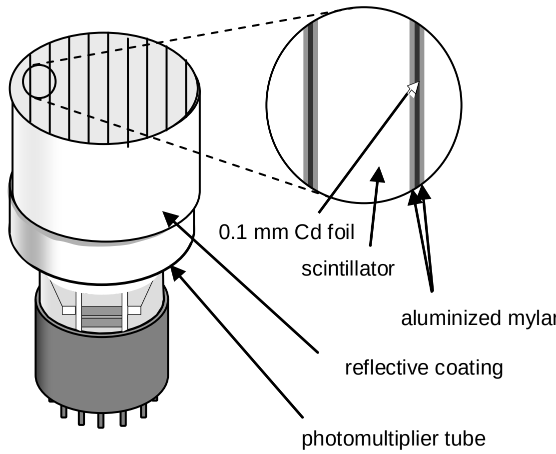

Fig. 2. Schematic of a novel cadmium-plastic scintillation capture-gated neutron detector.

3. Monte Carlo Analysis

3.1 MCNP-PoliMi Description

Monte Carlo codes have been widely used to design and analyze measurements such as those considered here; however, when modeling the time-correlated events resulting from gamma-ray interrogation, the widely-used Monte Carlo code MCNPX has some limitations. Specifically, MCNPX deviates from physical reality and the particles resulting from fission are not modeled correctly on an event-by-event basis [1]. A modified version of MCNP4C called MCNP-PoliMi has been developed to simulate time-correlated quantities and include a correlation between individual neutron interactions and corresponding gamma-ray production [3]. MCNP-PoliMi version 1.3.2 is capable of running with all standard MCNP source types and includes several specific spontaneous-fission-source distributions with neutron and gamma-ray source multiplicities (for example, ^{252}Cf, ^{240}Pu, and ^{242}Pu). However, there is no correlation between the number of emitted neutrons and gamma rays. MCNP-PoliMi also contains angular distributions for fission neutrons that were applied in these calculations.

MCNP-PoliMi produces a data file of all interactions in the detector cells that is read by a post-processing code to compute the detector-specific response. In the case of a scintillation detector, the incoming radiation must deposit enough energy to overcome a specific threshold for light output; this threshold is common for neutrons and gamma rays. Different incoming particles interact in different ways: gamma rays interact primarily through Compton scattering on electrons, while neutrons interact through scattering on hydrogen and carbon. The energy deposition in each interaction is converted into light in the post-processor using empirical relationships [4]. The event-by-event interactions modeled in MCNP-PoliMi enables the simulation of detailed detection physics, which is typically disregarded in other simplified code systems. The data given in the MCNP-PoliMi output file makes modeling effects such as varying light output of different target nuclei and multiple particle-scattering events possible.

Furthermore, MCNP-PoliMi imposes energy conservation on the Q-value of each individual interaction, while the standard-MCNP treatment does not. Figure 3 shows the results of a simulation in which a beam of thermal neutrons (E_n = 0.0253 eV) was incident on a 0.1-mm thick ^{nat}Cd sheet. The calculation was performed with the MCNP-PoliMi gamma-ray treatment and with standard-MCNP treatment. The energy spectrum of the photons exiting the sheet was tallied. The MCNP-PoliMi treatment differs significantly from the standard-MCNP treatment, specifically for photons greater than 2 MeV. The average energy of the PoliMi-treatment gamma rays is 2.649 MeV and the average of the standard-MCNP gamma rays is 2.462 MeV.

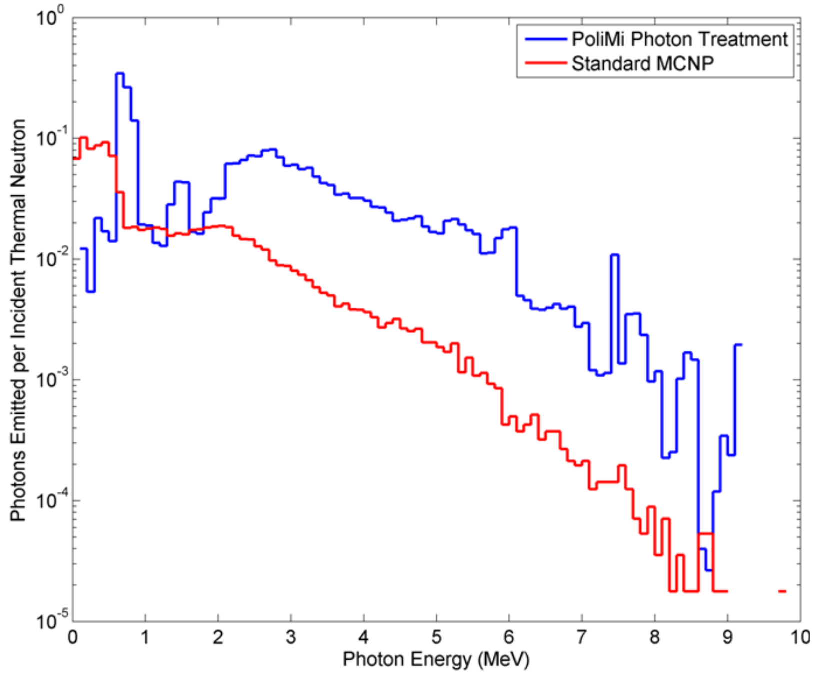

Fig. 3. Comparison of radiative-capture gamma-ray energy spectra with various physics treatments.

3.2. MCNP-PoliMi Detector Model

The MCNP-PoliMi model included only the active volume of the Cd-plastic capture-gated detector: the PMT and electronics were omitted for simplicity. The detector is a cylindrical active volume of 12 0.1-mm thick Cd foils in alternating layers with 13 1-cm thick BC-408 plastic scintillator slabs. The diameter and depth of the detector active volume are 13.28 cm and 15 cm. The composition of the BC-408 plastic scintillator is 52.5% hydrogen and 47.5% carbon, by atom, with a density of 1.032 g/cm^3. The cadmium foils were modeled as natural cadmium metal with a density of 8.650 g/cm^3. The ENDF-B/VI cross-section data were used for the BC-408 material, while the ENDL92 cross-section data were applied for the cadmium material. The reason for this was that that ENDL92 library is the only MCNP-distributed library that contains secondary gamma-ray-production data. The JEFF3.1 library, which is not included in the standard-MCNP distribution, also contains secondary gamma-ray-production data. These data have been acquired through the Radiation Safety Information Computational Center, and will be tested in the future.

The detector was simulated with a ^{252}Cf source placed 15 cm away in vacuum. The simulation configuration is shown in Figure 4. The simulations were performed using MCNP-PoliMi v. 1.3.2 and pulse height distributions were computed for the gamma rays that interacted in the BC-408 sheets. As described in Section 3.1, the simulation was performed using both the MCNP-PoliMi secondary-gamma-ray treatment and the standard-MCNP treatment for comparison. The gamma-ray pulse height distribution is shown in Figure 5.

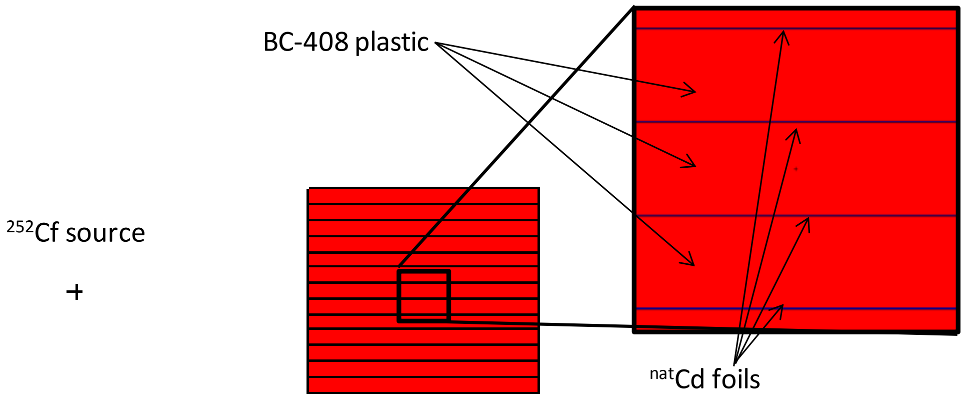

Fig. 4. MCNP-PoliMi simulation model of the Cd-plastic capture-gated scintillation detector with a ^{252}Cf source placed at 15 cm in vacuum.

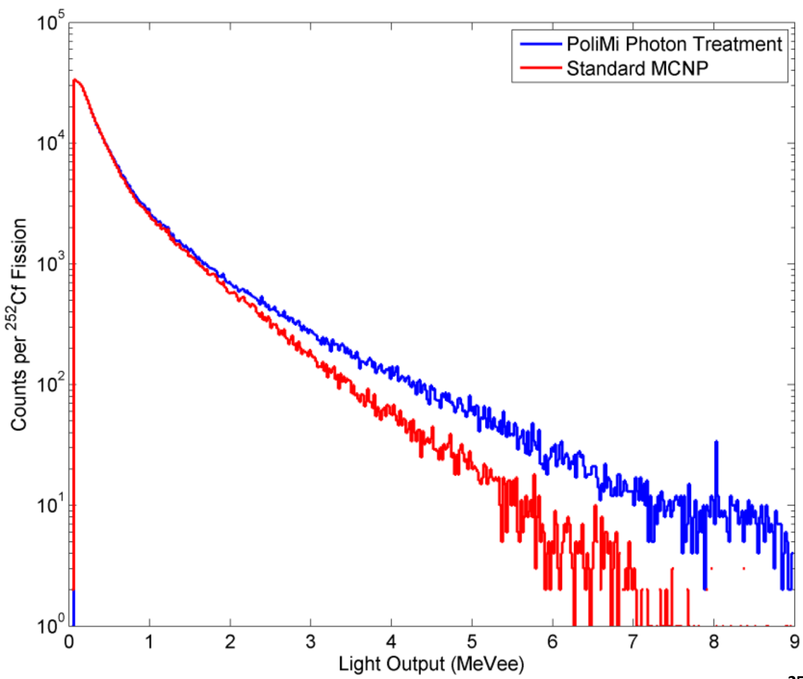

Fig. 5. Simulated pulse height distributions with varying physics treatment from a ^{252}Cf source 15 cm from the Cd-plastic capture-gated scintillation detector.

4. Measurement Analysis

4.1 DNNG-Developed Measurement System

Measurements were performed using the fast digital measurement system developed by Detection for Nuclear Nonproliferation Group (DNNG) at the University of Michigan. The basis of the system is a CAEN V1720 waveform digitizer shown in Figure 6. The V1720 is a 12-bit, 250-

MHz analog-to-digital converter with a 2-V dynamic range. There are eight independent channels on the V1720 board which allows for correlation and time-of-flight measurements with multiple detectors. Data from the V1720 are transferred to a computer with an optical cable controlled through DNNG-developed acquisition software. This allows for real-time data collection from fast organic scintillators at count rates greater than 70,000 counts per channel. The acquired pulse trains in a user-defined time window are written to a list-mode data file for subsequent analysis.

Fig. 6. CAEN V1720 waveform digitizer.

4.2 Measurement Results

A measurement was performed with a ^{252}Cf source placed 15 cm from the Cd-plastic capture-gated scintillation detector. Data were collected in list-mode format using the acquisition system described in Section 4.1. The data were analyzed using specialized capture-gating algorithms developed by the DNNG. The algorithms search through the list-mode data to identify any pulse trains containing multiple pulses within a user-specified acceptance time-window. For this work, the acceptance time-window was set to 20 μs. It is assumed that the primary pulse in these “capture-gated pulse trains” is from a scatter neutron and the secondary pulse is from capture gamma rays. Figure 7a shows the distribution of the time differences between the primary and secondary pulses in each acceptance time window. The average of this distribution is 4.73 μs. Figure 7b shows the distribution of the pulse heights of the primary and secondary pulses in each acceptance time window. The average of the primary-pulse distribution is 0.25 MeVee and the average of the secondary-pulse distribution is 0.41 MeVee. Because the Q-value of the reaction that produces the gamma rays is 9 MeV, it is expected that the secondary pulse height distribution is greater than the primary distribution: the primary pulses are from neutrons which are less energetic and produce less light per interaction than gamma rays. This is a promising measurement result because it indicates a relatively large difference between the pulse heights of primary and secondary pulses can be expected. This fact can be utilized by setting a certain pulse height threshold for the secondary pulses to minimize the number of falsely identified capture gamma rays. The most appropriate threshold will be investigated in the future.

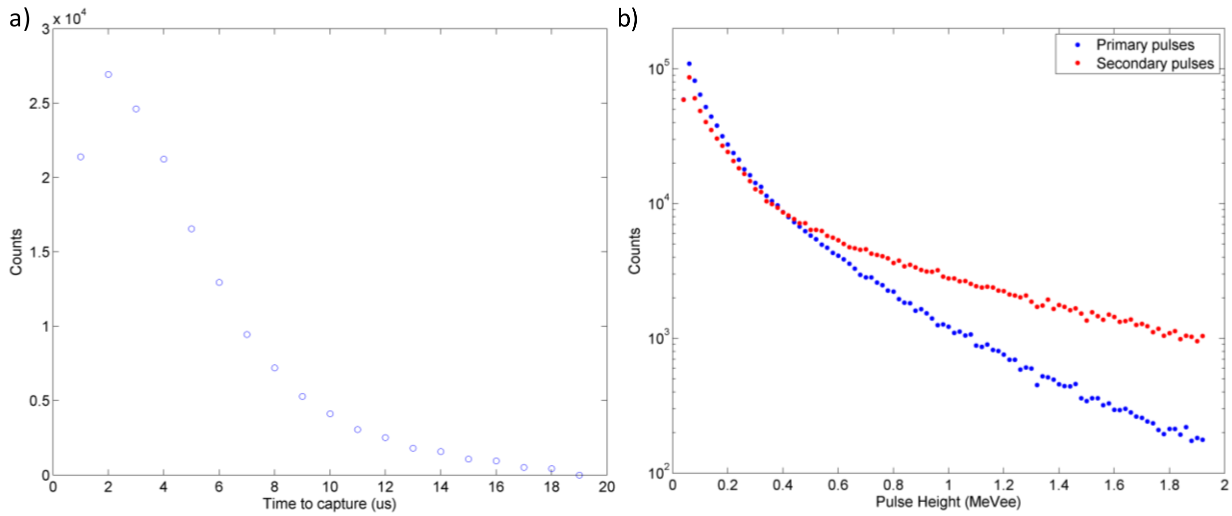

Fig. 7. Results from a ^{252}Cf source measured with the Cd-plastic capture-gated scintillation detector: a) time-to-capture distribution with a 20-μs acceptance time window; b) pulse height distributions of primary (scatter neutron) and secondary (capture gamma rays) pulses.

5. Conclusions

Capture-gated scintillators are being explored as a possible replacement technology for ^{3}He-based systems. Preliminary simulations and measurements have been performed with a novel capture-gated scintillation detector based on BC-408 plastic slabs and ^{nat}Cd foils. Simulations were performed using the MCNP-PoliMi code which provides energy conservation on an event-by-event basis. This is imperative because capture-gated detectors rely on the analysis of correlated particles from the same nuclear interaction: radiative capture in this case. The MCNP-PoliMi analysis differs from the standard-MCNP treatment. Measurements of the pulse heights of primary (scatter neutrons) and secondary (capture gamma rays) pulses have been made. A significant difference in average pulse height was observed and is encouraging because a threshold can be set on secondary pulses to minimize accidentals.

Acknowledgement

This research was funded by the National Science Foundation and the Domestic Nuclear Detection Office of the Department of Homeland Security through the Academic Research Initiative Award # CMMI 0938909.

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