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Reliability of Monte Carlo simulations in modeling neutron yields from a shielded fission source

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ABSTRACT

Using the combination of a neutron-sensitive ^{6}Li glass scintillator detector with a neutron-insensitive ^{7}Li glass scintillator detector, we are able to make an accurate measurement of the capture rate of fission neutrons on ^{6}Li. We used this detector with a ^{252}Cf neutron source to measure the effects of both non-borated polyethylene and 5% borated polyethylene shielding on detection rates over a range of shielding thicknesses. Both of these measurements were compared with MCNP calculations to determine how well the calculations reproduced the measurements. When the source is highly shielded, the number of interactions experienced by each neutron prior to arriving at the detector is large, so it is important to compare Monte Carlo modeling with actual experimental measurements. MCNP reproduces the data fairly well, but it does generally underestimate detector efficiency both with and without polyethylene shielding. For non-borated polyethylene it underestimates the measured value by an average of 8%. This increases to an average of 11% for borated polyethylene.

1. Introduction and motivation

Currently, radiation portal monitoring (RPM) systems are used to help prevent the spread of special nuclear materials. Neutron detectors are an integral part of these systems since they can detect the presence of nuclei such as ^{239}Pu, which have high spontaneous fission rates. It is important that these detectors are sensitive not only to bare fission sources, but also to shielded sources because illicitly imported material will likely be shielded to avoid detection. Kouzes et al. [1] performed extensive MCNP (Monte Carlo Neutral Particle) [2] simulations of the response of moderated ^{3}He detectors to fission sources embedded in a variety of shielding materials. Rees and Czirr [3] extended these simulations to include several moderating strategies for ^{3}He tubes detecting neutrons emitted from fission sources shielded with polyethylene. Tomanin et al. [4] along with Gilbert et al. [5] have done additional computational studies to optimize ^{3}He detectors for national security applications. When MCNP simulations are used to model shielded sources, they often require the modeling of multiple neutron interactions. Small discrepancies in cross sections or modeling techniques could possibly lead to significant deviations from experimental data. An experimental measurement of neutron flux from shielded sources is important as a check of the validity of such simulations.

The results of such experiments are dependent on the shielding and the detectors that are used. Polyethylene is an inexpensive, efficient, and readily available neutron moderator and is used in many neutron shielding applications because of its high hydrogen content. High hydrogen content is desirable because neutron–hydrogen collisions cause neutrons to lose an average of about half of their energy per collision. To improve the absorption of neutrons in the polyethylene, boron, which has a very high neutron capture cross section, can be added. Borated polyethylene has been found to be a considerably better shielding material [6] than non-borated polyethylene. We constructed a moderated lithium-glass, Fig. 1, that is sensitive to both high and low energy fission neutrons. Using this detector, we measured the effects of shielding a ^{252}Cf fission source with both non-borated polyethylene and borated polyethylene.

2. Experimental methods

2.1. Detectors

The choice of detector in this measurement is important as the results of both the experiment and the simulation are very detector dependent. The following criteria were the most important:

1) Since the energy of neutrons from shielded sources can vary from thermal to over 10 MeV, the detector should be sensitive

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0168-9002/

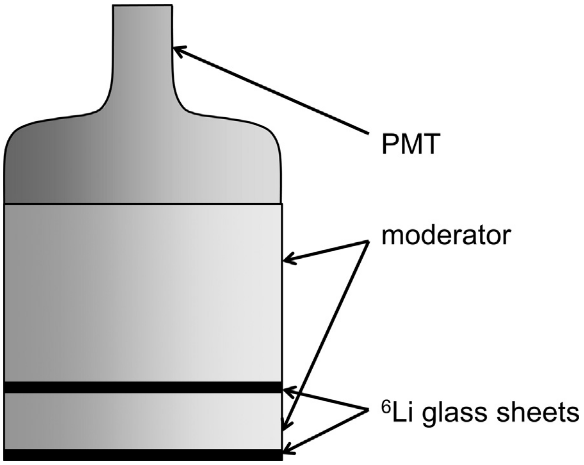

Fig. 1. Detector geometry.

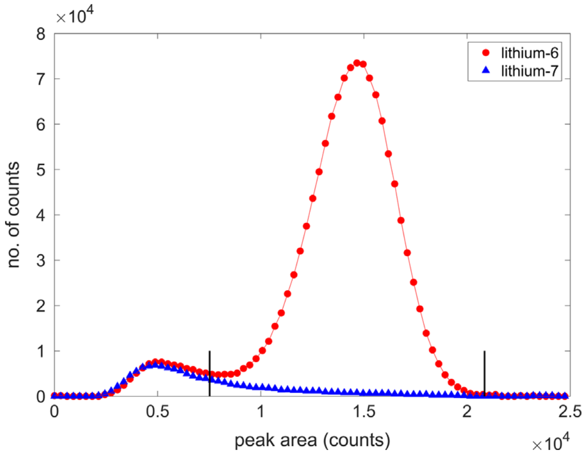

Fig. 2. Distribution of pulses in the ^{6}Li (circle) detector and the ^{7}Li (triangle) detector as a function of pulse area. The data are for 3.0 cm of non-borated polyethylene shielding above and below the source.

to a wide range of neutron energies.

2) The signals produced by low-energy neutrons should not be small enough to be confused with noise.
3) The detector should produce one pulse for every neutron that MCNP would count as detected. If a simulation has to calculate light output and light transmission to determine if a pulse reaches a threshold value, it is much less likely to be reliable.
4) The detector must be able to distinguish neutrons from gammas either event by event or in the aggregate.

A variety of detectors could meet these criteria. A ^{3}He detector placed in a box with 5 cm-thick polyethylene walls, a typical moderator [1], does not meet the first criterion as it has little sensitivity to thermal neutrons [3]. Organic scintillators produce energy-dependent signals, and where neutron-gamma discrimination is possible, the discrimination is not good when pulse heights are small. Our choice was to build detectors with lithium glass and lucite moderators. ^{6}Li has a large cross section for neutron capture and has been used in a number of neutron detector applications including glass fibers [7], liquid scintillators [8], plastic scintillators [9], and lithium glass [10].

Two neutron detectors were used in conjunction with each other for this study. The first consisted of two sheets of ^{6}Li-doped glass scintillator (Applied Scintillation Technologies GS-20, lithium is 95% ^{6}Li), each 1 mm thick and 12.7 cm in diameter. The first piece of glass was placed at the front of the detector followed by a 2.5 cm-thick lucite moderator. Lucite was chosen because it has a reasonably large hydrogen content, it is optically clear, it is easily machined, and it is available in convenient sizes. The second sheet of glass scintillator was placed on top of the lucite and then an additional 7.6 cm-thick lucite moderator was added. The glass and lucite all have 12.7 cm diameters. Finally, we place a 12.7-cm diameter Adit Model B133D01S photomultiplier tube on top of the second lucite moderator, as shown in Fig. 1. The pulses of the photomultiplier tube are recorded by a signal digitizer so we can restrict the analysis to pulses with areas consistent with neutrons.

The second detector was identical to the first, except that the ^{6}Li glass was replaced by ^{7}Li glass (Applied Scintillation Technologies GS-30, lithium is 99.99% ^{7}Li). The sheets of GS-20 and GS-30 are identical except for the isotopic composition of the lithium. The ^{7}Li detector is used to remove ambient and source-related gamma backgrounds from the neutron signal, as described below. This technique was used because the ^{6}Li detector is incapable of clean event-by-event neutron-gamma discrimination.

The ^{6}Li in the glass scintillator has a large capture cross section for lower-energy neutrons. The neutron capture and subsequent fission process is as follows:

$$n + {}^6\text{Li} \rightarrow \alpha(2.056\ \text{MeV}) + {}^3\text{H}(2.728\ \text{MeV}) \qquad (1)$$

The alpha particle has a range of 6.2 μm in the lithium glass and the triton has a range of 35.2 μm. Therefore, both particles will generally deposit all of their energy in the 1 mm-thick piece of glass. The intensity of the light pulse produced by the scintillator in the glass is constant for all detected neutrons. This is because the neutron kinetic energy is typically very small for capture events so that the final-state charged particles always have essentially the same energies. There is a little variation in signal size, but this is more a matter of optical considerations. This can be seen in Fig. 2, a histogram of the peak areas for both ^{6}Li and ^{7}Li detectors, where the neutron peak from ^{6}Li is seen clearly on the right.

Two sheets of glass scintillator are used because the incident neutrons can vary greatly in energy. Thermal neutrons will almost all be absorbed by the first sheet of glass. Most neutrons with energies of roughly 100 keV or higher pass into the lucite where they are moderated. Once moderated, these neutrons can be captured in either piece of glass. Monte Carlo simulations [3] show that the neutrons usually lose most of their energy within two or three collisions and then undergo a random walk within the moderating material. This effectively produces a “neutron cloud” within the moderator. The depth of the cloud depends on the energy of the incident neutrons. We place the second sheet of glass scintillator at a depth that is optimized for 250 keV neutrons. This energy is lower than the peak of the energy distribution for a bare source, but is a good choice for neutrons from a shielded source. An MCNP simulation of the relative efficiency of each sheet of glass for absorbing neutrons is shown in Fig. 3. The shielding thickness reported throughout the paper is the thickness of the shielding on either side of the source, so the total thickness is twice that amount. The neutrons from the unshielded source have sufficiently high energy that 70% of all neutrons captured in lithium glass are captured in the second glass sheet; however, as shielding surrounding the source increases and the energy of the neutrons leaving this shielding decreases, nearly all of the neutron capture occurs in the first sheet of lithium glass. This allows for high sensitivity over a broader range of neutron energies. A Monte Carlo simulation suggests that the efficiency for the simple detector shown in Fig. 1 (with no can) to normally incident neutrons is 66% for 1-eV neutrons, 17% for 100-keV neutrons, 7.8% for 1-MeV neutrons, and 1.1% for 10-MeV neutrons.

While these detectors work well in the specific application for which they were designed, they are not suitable for portal