Time-Dependent Simulation of Neutron Detector Response

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1.0 INTRODUCTION

Neutron time-of-flight (TOF) measurements are applicable to nonproliferation efforts to indentify special nuclear material (SNM). However, current Monte Carlo codes such as MCNPX cannot simulate neutron detector response [1]. MCNP-PoliMi and its detection postprocessor can simulate neutron detector response, but only if the simulation is performed in analog mode, resulting in time-consuming simulations [2, 3]. This technique has been applied in the past and has shown good agreement with measured data [4, 5].

In this paper, the TOF distribution from a ^{252}Cf source is computed using a novel method and compared to measured results from the Detection on Nuclear Nonproliferation (DNNG) Lab at the University of Michigan. Monte Carlo techniques are used to compute the neutron spectrum entering the detector and a predetermined detector response matrix is then used to predict the distribution of the detected neutrons as a function of time.

2.0 THEORY

2.1 Geometry Description

The measurement set-up contains an isotropic ^{252}Cf source placed 40 cm from the face of an EJ-309 liquid scintillation detector. The detector is a cylinder with a radius of 6.34 cm and a length of 12.51 cm. The chemical composition is 54.8% hydrogen and 45.2% natural carbon by number. It is assumed that neutrons enter the detector uncollided. This is reasonable because a very small fraction of neutrons scatter in air before reaching the detector in case of a 40 cm source-detector distance. The 40 cm source-detector distance subtends a maximum angle of 9° along which a particle can travel. This justifies the second assumption: particles incident on the detector face are mono-directional. The last assumption is that the first scatter of a neutron defines the timing of the light pulse, and only one light pulse is created per incident neutron.

2.2 Sampling Procedure

In order to simulate the ^{252}Cf neutrons detected by the EJ-309 detector three parameters are randomly sampled: the energy of the neutrons, the radial distance from the center of the detector face, and the distance travelled before the first collision after being incident. The velocity of the neutron is determined by its energy which is sampled from the MCNP-PoliMi ^{252}Cf internal source distribution. To sample the radial distance at which the neutron is incident, r the cumulative density function (CDF)

$$\xi_1(r) = \int_0^r \frac{2\pi r'}{\pi R^2} dr' \tag{1}$$

is used for the detector radius, R. In order to sample distance travelled by the neutron before its first collision the following CDF is used:

$$\xi_2(x) = \int_0^x \Sigma_T^{H+C}(E) * e^{-\Sigma_T^{H+C}(E) * x'} dx' \tag{2}$$

Here $\Sigma_T^{H+C}(E)$ is the total macroscopic cross-section at an energy, E for the EJ-309 detector material, x is the distance travelled until the first collision after the neutron is incident on the detector. Given the randomly sampled energy and the total distance travelled by the neutron (calculated using randomly sampled parameters r and x) the TOF detector response is determined. Each neutron was tallied based on its weight which is the intrinsic efficiency that a neutron incident at that energy will be detected. The intrinsic efficiencies have been extracted from an EJ-309 detector response matrix simulated with the MCNP-PoliMi code.

3.0 RESULTS & ANALYSIS

A comparison of the simulated TOF distribution with measured TOF distribution is shown in Figure 1. The agreement in the distributions are good, however, it can be noticed that the simulated TOF distribution is skewed towards longer detection times. Work continues on resolving this discrepancy. The measured result has an unusual feature at about 6 ns that is suspected to be contribution from the gamma peak that has been excluded.

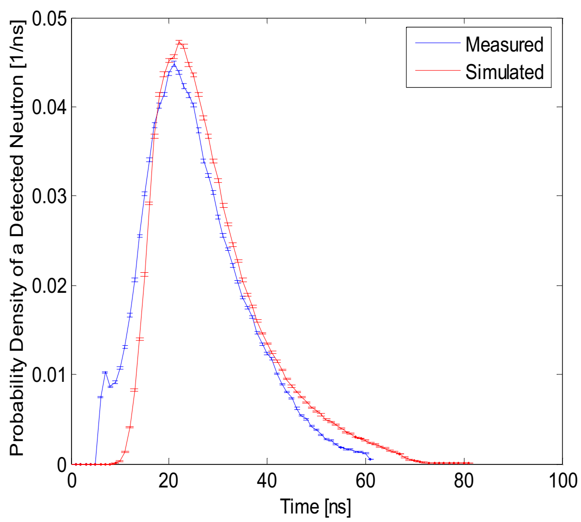

Figure 1: A comparison of the TOF distributions [6]

4.0 CONCLUSION

TOF measurements have several applications in the field of nonproliferation. They can be important in characterizing fissionable sources. However, they are difficult to simulate using standard Monte Carlo codes, which cannot simulate neutron detector response. A novel technique has been applied here to simulate TOF detector response to an unshielded ^{252}Cf source. This was done be randomly sampling neutron parameters such as its energy and its flight path and solving for the TOF. An intrinsic efficiency response matrix for the EJ-309 detector was then utilized to tally for the detected neutrons only. Preliminary result show good agreement with the measured results from the DNNG lab. Future work will include investigation of the slight shift of the simulated results to longer detection times. This method will then be applied to shielded configurations and for varying sources.

5.0 REFERENCES

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