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Validation of the MCNPX-PoliMi Code to Design a Fast-Neutron Multiplicity Counter

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Abstract
The University of Michigan's Detection for Nuclear Nonproliferation Group (DNNG) in collaboration with Idaho National Laboratory is developing a fast-neutron multiplicity counter with organic liquid scintillators to measure important nuclear-safeguard quantities such as plutonium mass. The system is being designed and optimized with simulations performed using the MCNPX-PoliMi particle transport code. Validation measurements have been performed at the University of Michigan and the Joint Research Centre in Ispra, Italy to justify the use of the Monte Carlo code as well as the DNNG's MPPost multiplicity routine. Results from these validation measurements are presented in this paper.

1. Introduction

As the world's energy demands escalate, the drive to expand the use of nuclear energy continues to be a priority all around the world. This can only be accomplished in a safe way if nuclear fuels are adequately safeguarded. Robust methods to accurately account for various quantities of special nuclear material (SNM) of varying type, size, and shape must be developed and tested to allow regulators to confidently move forward in the expansion of their nuclear-energy-production activities.

Many safeguards measurement systems used at nuclear facilities, both domestically and internationally, rely on He-3 detectors and well-established mathematical equations to interpret coincidence and neutron-multiplicity measurements for verifying quantities of SNM. Due to resource shortages alternatives to these existing He-3 based systems are being sought. Work is also underway to broaden the capabilities of these types of measurement systems in order to improve current multiplicity analysis techniques.

As a part of a Material Protection, Accounting, and Control Technology (MPACT) project within the U.S. Department of Energy's Fuel Cycle Technology Program, the Detection for Nuclear Nonproliferation Group (DNNG) is designing a fast-neutron multiplicity counter with organic liquid scintillators to quantify important nuclear-safeguards quantities such as plutonium mass. In addition to the system design, this project aims to quantify the novel design's performance and examine the potential benefits of using fast-neutron detectors for multiplicity analysis of advanced fuels in comparison with He-3 detectors. The designs are being developed and optimized using the MCNPX-PoliMi transport code [1] to study detector response.

In this paper, we discuss validation measurements used to justify the use of the MCNPX-PoliMi code paired with the DNNG's MPPost [2] multiplicity routine, to design a fast neutron multiplicity counter with liquid scintillators. This multiplicity counter will be designed with the goal to safeguard advanced nuclear fuels. With improved timing qualities associated with liquid scintillation detectors, we are convinced that the new system, due to its fast time response, will be less prone to signal overlap typical for nuclear materials with high radiation activities.

2. Fast-Neutron Multiplicity Experiments

To benchmark the MCNPX-PoliMi code and the MPPost post-processor, measurements were performed in Ispra, Italy using a well-characterized Cf-252 source and a sample of mixed-oxide (MOX) powder surrounded by a 1-cm thick tube of lead. The composition of the MOX is outlined in Fig. 1.

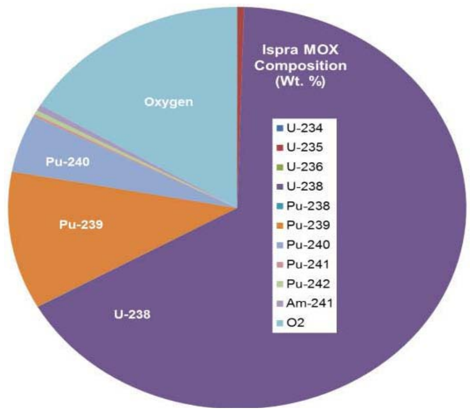

Figure 1. MOX-material composition.

Four 7.62-cm x7.62-cm EJ-309 organic liquid scintillator detectors were used with a CAEN DT5720 digitizer to collect multiplicity and pulse-height data. A 3.4 μCi Cf-252 source was placed 40 cm from the center of each detector face, with 12.5 cm between the detectors as shown in Fig. 2. The four liquid scintillators represent one third of a full detector ring present in a potential prototype. The MOX sample was measured twice—once in the same configuration as the Cf-252 source, with the addition of 1-cm of lead shielding, and once with a three-sided polyethylene box (8-cm thick) and lead shielding (hereafter referred to as the moderated case). The moderated case setup is shown in Fig. 3.

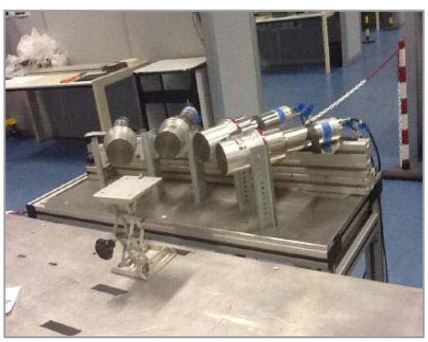

Figure 2. Cf-252 experimental setup.

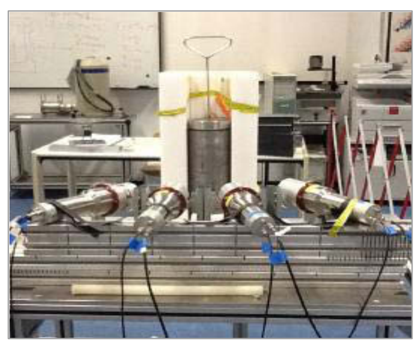

Figure 3. Moderated-MOX experimental setup.

3. MCNPX-PoliMi Simulations

The measurements were simulated with the Monte Carlo code MCNPX-PoliMi for the radiation transport and the MPPost post-processor to convert the radiation transport details into a measurable detector response. The simulated moderated case is shown in Fig. 4. MCNPX-PoliMi has the improved capability of simulating correlated source events so that multiplicity can be analyzed. MPPost takes the detailed particle interaction data file from MCNPX-PoliMi and produces pulse-height and neutron-multiplicity distributions based on the specifications of the measurement system. This project focused only on neutron analysis, although the simulation package provides photon analysis as well.

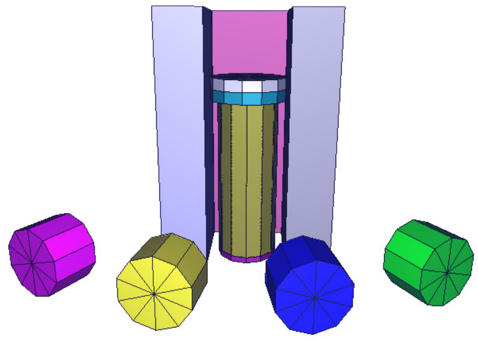

Figure 4. MCNPX-PoliMi model of the moderated-MOX case.

The pulse-height distribution is created by converting the energy deposited in each event in the detectors to a pulse height in units of light output. These pulse heights are then be histogrammed into a specified binning scheme, 10-keVee wide bins in this case. The neutron-multiplicity is recorded first by creating a list of accepted pulses, i.e. not lost to dead time, pile-up rejections, or the digitizer threshold of 70 keVee. Each neutron detection that "triggers" the digitizer opens a 400-ns window for the measurement of a double or triple neutron event. In reality, the acceptance time window is less than 400 ns when pulse-shape discrimination algorithms are considered.

Simulating the MOX source in MCNPX-PoliMi is done by modeling the anticipated neutron emission from individual isotopes present in the mixture. For plutonium isotopes, PoliMi incorporates neutron and photon source distributions for both the spontaneous fission and the alpha-n reactions (due to the presence of oxygen). Based on the material composition and the specific activity of these sources, the MOX source term can be defined.

4. Results

Figures 6 through 11 show the results comparing the simulated and measured neutron-multiplicity and pulse-height distributions for the three experimental configurations. The multiplicity results showed very good agreement in the MOX cases and reasonably good agreement in the Cf-252 case. The percent differences are shown in Table 1.

The pulse-height distributions follow correct trends and vary most significantly at higher energies where poor statistics again become a factor. The simulated Cf-252 pulse-height distribution under predicts in the low energy region which is likely due to missing scattered neutrons, indicating that the addition of more geometric detail could improve the comparison.

The triples percent difference is inconsistent because of the poor statistics, with fewer than 10 counts in an hour in all cases. The doubles and singles show very good agreement, however.

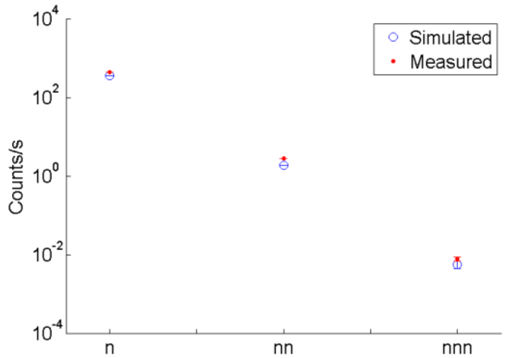

Figure 6. Comparison of simulated and measured Cf-252 neutron-multiplicity distributions.

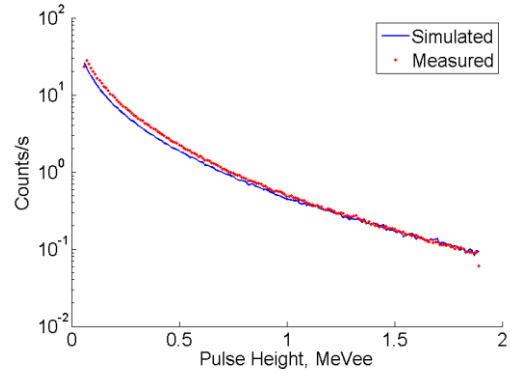

Figure 7. Comparison of simulated and measured Cf-252 pulse-height distributions.

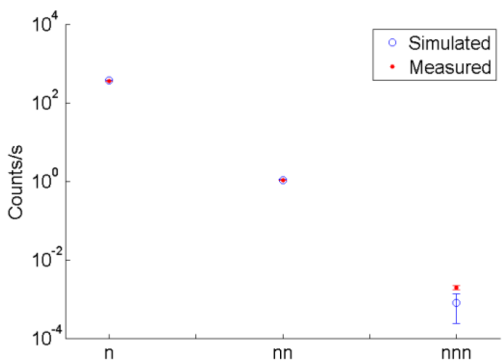

Figure 8. Comparison of simulated and measured MOX neutron-multiplicity distributions.

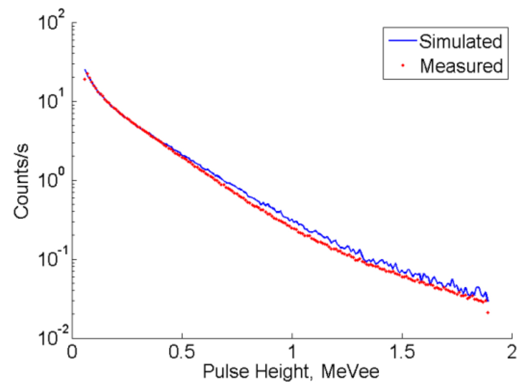

Figure 9. Comparison of simulated and measured MOX pulse-height distributions.

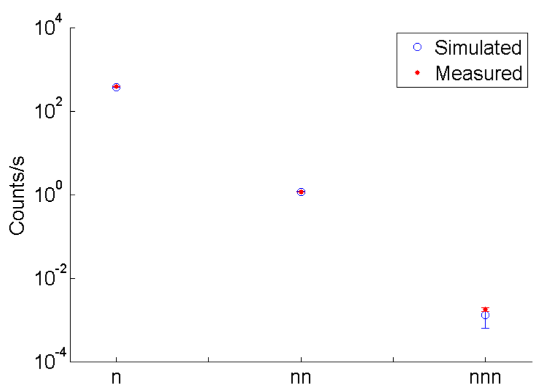

Figure 10. Comparison of simulated and measured moderated MOX neutron-multiplicity distributions.

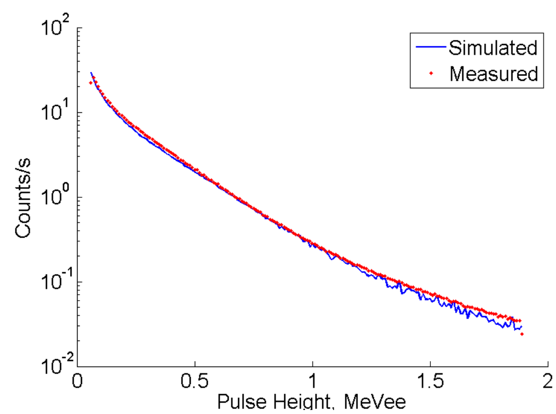

Figure 11. Comparison of simulated and measured moderated MOX pulse-height distributions.

Table 1. Percent differences between simulated and experimental neutron-multiplicity data.

	n	nn	nnn
Bare Cf-252	-17.45%	-32.29%	-27.59%
MOX w/ Pb	-17.40%	-19.98%	-67.32%
Moderated MOX w/ Pb	-5.88%	2.10%	-27.67%

5. Conclusions and future work

The purpose of this work was to validate the MCNPX-PoliMi code and the MPPost multiplicity routine in simulating a fast-neutron multiplicity counter. As shown, MCNPX-PoliMi and MPPost are successful in simulating the measured neutron-multiplicity and pulse-height distributions. The percent differences between the measured and simulated results are reasonable and justify the use of these programs to evaluate the feasibility of a fast-neutron multiplicity counter using liquid scintillators to measure important nuclear-safeguards quantities such as plutonium mass. These validations show the importance of detailed geometry in the simulations and future work will include an analysis of the sensitivity of the data analysis to geometric details in the simulations. In furthering the design of the multiplicity counter, additional work will be done to expand the simulations and measurements to full rings of detectors and optimization of sample-to-detector distance and detector spacing will be performed.

6. Acknowledgements

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7. References

[1] E. Padovani, S. A. Pozzi, S. D. Clarke, E. C. Miller, "Introduction to MCNPX-PoliMi", Version 2.7, (2012).
[2] E. C. Miller, S. A. Pozzi, E. Padovani, S. D. Clarke, M. Flaska, J. Mattingly, "MPPost Manual," Version 2.7, (2012).