LA-UR-13-24756
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Title: Comparing Radiation Detection Experiments to Simulation with MCNP

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Intended for: UCAL President Visit to LANL, 1 July 2013

Issued: 2013-06-27

Comparing Radiation Detection Experiments to Simulation with MCNP

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LANL XCP-7

Introduction

LANL has long been a leader in the simulation of radiation interaction with matter (a.k.a. radiation transport). The Monte Carlo N Particle (MCNP) radiation transport code, developed in LANL's XCP division, is one of a few transport codes developed at LANL. MCNP is used throughout the world for criticality safety, shielding, medical physics, and many other applications.

While transport codes can simulate how radiation particles, e.g., neutrons, γ-rays, x-rays, β particles, etc., will interact in a specified geometry, the results they produce are only as good as the underlying data that describes how many and what kinds of interactions that a particle will have. The United States maintains a set of nuclear data used in radiation transport codes called the Evaluated Nuclear Data Files (ENDF). Data from ENDF are processed into formats that can be utilized by radiation transport codes, such as MCNP.

Verifying Code Output with Experiment

Evaluation of the quality of codes and the underlying data is done by comparing simulations using that data to experiments. Proper simulation of a benchmark experiment requires that every aspect of the experiment affecting the result be properly simulated. For most radiation transport simulations this requires that the source of radiation particles, the geometry and materials in which the radiation particles are interacting, and the manner in which a detector responds to those particles must all be correctly modeled.

Neutron Multiplication Comparisons

Benchmark measurements of the BeRP ball, a sphere of plutonium, were recently made by a researcher at SNL. The data collected in the benchmark experiment included neutron multiplication measurements, neutron rate measurements and gamma-ray spectra measurements. Simulations of their experiments have been performed with MCNP with particular interest in the neutron multiplication. As polyethylene reflectors are added to exterior of the BeRP ball more neutrons are reflected that in turn produce more fissions, and thus giving the system a higher neutron multiplication. Figure 1 shows a rendering of the model geometry for these experiments.

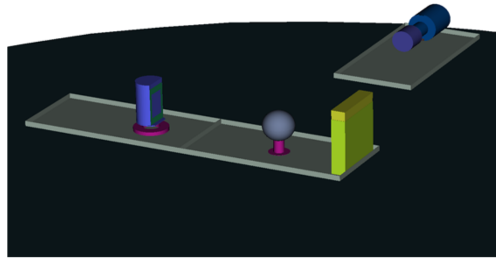

Fig. 1. Modeled BeRP Ball Experimental Measurement Configuration

Simulation of the benchmark experiment with MCNP showed a discrepancy between the simulated and the experimental multiplication curves, which plot the probability of detecting a given number of neutrons in coincidence. Figure 2(a) illustrates this discrepancy. By studying energy dependent perturbations to the average number of neutron emitted per fission, new values for the average number of neutrons per fission could be found that are within the uncertainties of the ENDF data and better reproduce this experimental data while preserving agreement with criticality experiments. Further work and evaluation is underway to determine if the new values should replace the current evaluation.

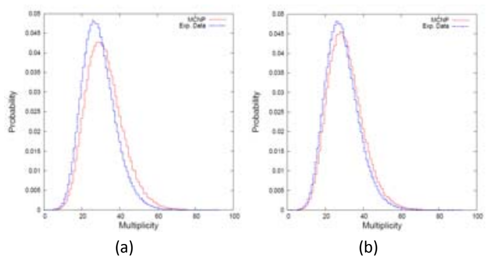

Fig. 2. Comparison of multiplication curves obtained using (a) ENDF data and (b) modifications to the ENDF data within uncertainties that improve agreement to benchmark measurements

Gamma- and X-Ray Spectra Comparisons

Measurements of gamma- and x-ray spectra have also been compared to calculations with MCNP. The source particle emissions are dependent on the isotopic composition of the source material. The MCNP Intrinsic Source Constructor (MISC) code is used to generate the emission probabilities of each radiation particle.

Figure 3 presents a comparison of dose rates from a sphere of depleted uranium (DU) measured with a NaI detector with an MCNP calculation. The gamma-ray emission source terms for MCNP were calculated using MISC with older ENDF version 6 data and newer ENDF version 7 data. The ENDF version 7 data shows significant improvement over the version 6 data for the emission line at 1 MeV.

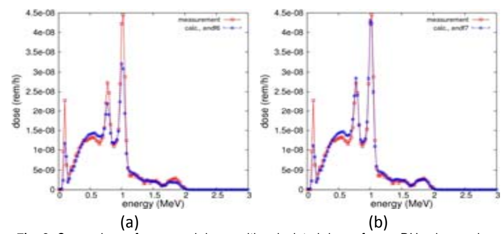

Fig. 3. Comparison of measured doses with calculated doses from a DU sphere using (a) ENDF version 6 and (b) ENDF version 7

Figure 4 shows the comparison of a measured ^{137}Cs spectrum from a HPGe detector and an MCNP simulation of the measurement. Again, MISC was used to generate the source description for MCNP with the ENDF version 7 data. The simulation is slightly lower than the experimental data and work is ongoing to understand why.

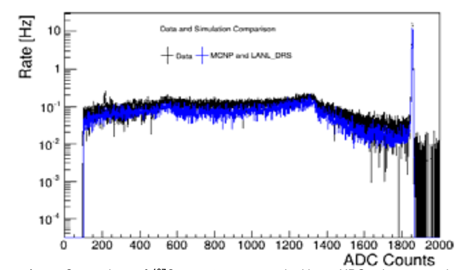

Fig. 4. Comparison of ^{137}Cs spectra measured with an HPGe detector and calculated with MCNP using ENDF version 7 data

Conclusions

Simulation of radiation detection experiments and benchmarks provides useful insight into the quality of the underlying nuclear data used by transport codes like MCNP. Where potential improvements to the nuclear data might be made can be studied by simulation and comparison to experiment.

References

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