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Validating the performance of correlated fission multiplicity implementation in radiation transport codes with subcritical neutron multiplication benchmark experiments

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ABSTRACT

Historically, radiation transport codes have uncorrelated fission emissions. In reality, the particles emitted by both spontaneous and induced fissions are correlated in time, energy, angle, and multiplicity. This work validates the performance of various current Monte Carlo codes that take into account the underlying correlated physics of fission neutrons, specifically neutron multiplicity distributions. The performance of 4 Monte Carlo codes - MCNP®6.2, MCNP®6.2/FREYA, MCNP®6.2/CGMF, and PoliMi - was assessed using neutron multiplicity benchmark experiments. In addition, MCNP®6.2 simulations were run using JEFF-3.2 and JENDL-4.0, rather than ENDF/B-VII.1, data for ^{239}Pu and ^{240}Pu. The sensitive benchmark parameters that in this work represent the performance of each correlated fission multiplicity Monte Carlo code include the singles rate, the doubles rate, leakage multiplication, and Feynman histograms. Although it is difficult to determine which radiation transport code shows the best overall performance in simulating subcritical neutron multiplication inference benchmark measurements, it is clear that correlations exist between the underlying nuclear data utilized by (or generated by) the various codes, and the correlated neutron observables of interest. This could prove useful in nuclear data validation and evaluation applications, in which a particular moment of the neutron multiplicity distribution is of more interest than the other moments. It is also quite clear that, because transport is handled by MCNP®6.2 in 3 of the 4 codes, with the 4th code (PoliMi) being based on an older version of MCNP®, the differences in correlated neutron observables of interest are most likely due to the treatment of fission event generation in each of the different codes, as opposed to the radiation transport.

1. Introduction

Experts in the fields of nuclear nonproliferation, safeguards, and criticality safety have been continually performing subcritical special nuclear material (SNM) measurements since the 1940s. The results of these experiments have provided data used for simulations of SNM systems. Improvements in nuclear detection instrumentation and SNM availability in the 1950s and 1960s lead to increased research activity in both the theory and practice of multiplication and reactivity measurements. Neutron multiplication is an extremely important parameter in SNM systems, as it can give information about the type, enrichment, and risk level of the SNM being investigated for nuclear security reasons. In addition, for criticality safety purposes, it is extremely important to be able to accurately predict the multiplication of systems for various processes and experiments. Neutron multiplication inference measurements take advantage of the fact that neutrons emitted during fission are correlated in time and can be used to gain knowledge about the system being measured. Multiplying system parameters of interest include neutron leakage multiplication M_L, total neutron multiplication M_T, the neutron multiplication factor k_{eff}, and the prompt neutron multiplication factor k_p. M_L represents the average number of prompt neutrons escaping a system for every neutron injected into the system, while M_T represents the number of prompt neutrons created on average by a single neutron in the multiplying system. k_{eff} is a measure of the ratio of the total number of neutrons in the current fission generation to the total number of neutrons in the previous generation. k_p is a measure of the ratio of the number of prompt neutrons in the current fission generation to the number of prompt neutrons in the previous

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generation. Some subcritical inferred neutron multiplication parameters of interest are sensitive to the distribution of the number of neutrons emitted per fission. Comparisons between subcritical neutron multiplication inference measurements and simulations have been used to validate multiplication inference techniques and radiation particle transport codes, and to identify and correct deficiencies in underlying nuclear data quantities such as $\bar{\nu}$ (average number of prompt neutrons emitted per fission) (Arthur et al., 2016; Bahran et al., 2014; Sood et al., 2014; Bolding and Solomon, 2013; Miller et al., 2010; Mattingly et al., 2009; Bahran et al., 2014; Boldeman and Hines, 1985). Most notably, recent (1990s and 2000s) methods of obtaining list mode data (time stamps of neutron events registered in a detector) from both measurements and simulations have also been developed and allow for a more detailed comparison between the two (Hutchinson et al., 2016).

More recently, there has been significant progress on the design and execution of benchmark quality subcritical neutron multiplication measurements for radiation transport code and nuclear data validation. The majority of these experiments have involved a 4.5 kg alpha-phase plutonium sphere (BeRP ball) surrounded by copper (Bahran and Hutchinson, 2016), tungsten (Richard et al., 2016), and nickel (Richard et al., 2016). The International Criticality Safety Benchmark Evaluation Project (ICSBEP) Handbook (Briggs, 2014) includes accepted evaluations of both the nickel and tungsten measurements. The ICSBEP handbook contains thousands of critical and subcritical measurement benchmark evaluations. The purpose of the handbook is to provide trusted benchmarks for validation and improvement of nuclear databases and radiation transport codes. The nickel benchmark was the first ICSBEP-accepted evaluation of measurements analyzed with the Hage-Cifarelli formalism based on the Feynman Variance-to-Mean method (Cifarelli and Hage, 1986), and was the culmination of many years of collaborative subcritical experiment research (Arthur et al., 2016; Bahran et al., 2014; Sood et al., 2014; Bolding and Solomon, 2013; Miller et al., 2010; Mattingly et al., 2009; Hutchinson et al., 2016; Richard et al., 2016; Richard et al., 2016; Hutchinson et al., 2013; Hutchinson et al., 2014; Hutchinson et al., 2013; Hutchinson et al., 2015).

This work investigates the performance of various current Monte Carlo codes that take into account at least some of the correlated physics of fission neutrons (i.e. correlations in time, energy, angle, or some combination of the three). Historically, radiation transport codes have uncorrelated fission emissions. In reality, both spontaneous and induced fissions release particles that are correlated in time, energy, angle, and multiplicity. The fission process can be either spontaneous or initiated by an interacting neutron. In the case of spontaneous fission, the nucleus is inherently unstable and randomly decays by fission. In the case of neutron-induced fission, an unstable compound nucleus forms after an incident neutron collides with the original nucleus. In either case, the nucleus scissions into two fission fragments, which receive some of the energy liberated from the rearrangement of mass as kinetic energy. The fission fragments release the remaining energy in the form of prompt neutron emission, prompt gamma ray emission, and delayed β or electron conversion decay. Because the particles are emitted from moving fission fragments, the multiplicities, energies, and angles of emission of prompt neutrons and gamma rays are dependent upon both each other and the initial masses and kinetic energies of the fission fragments (Wagemans, 1991). For this work, only prompt fission neutrons are of interest and the authors do not consider the physics of gamma production in fission. Because of their large impact on correlated neutron results, this work also compares underlying fission neutron multiplicity distributions utilized by the different codes.

2. Correlated fission multplicity

2.1. Nuclear data

For the purposes of this paper we will be focusing on the nuclear reaction database utilized by the general purpose Monte Carlo code MCNP®6.2[1], namely the Evaluated Nuclear Data File (ENDF) (Chadwick et al., 2011), although results will also be obtained using the Joint Evaluated Fission and Fusion File (JEFF) (Santamarina et al., 2009) and the Japanese Evaluated Nuclear Data Library (JENDL) (Shibata et al., 2011). ENDF contains information related to the types and probabilities of the different possible reactions between radiation particles and various isotopes. Evaluators use data from high-quality differential measurements to evaluate nuclear data libraries such as ENDF, and comparisons of simulated and measured data from benchmark-quality integral measurements to validate the libraries. Fig. 1 summarizes the process. Included in the information provided by ENDF are data summarizing both the probability of fission occurring and the average number of neutrons released per fission of each fissionable isotope, represented as $\bar{\nu}$, as functions of incident neutron energy. The multiplicity distribution $P(\nu)$ represents the probability for ν neutrons to be emitted per fission. Complete multiplicity distributions, $P(\nu)$, are not included in ENDF/B-VII.1; correlations in angle and energy are also not included.

Overall, the ENDF evaluation process focuses on complying as closely as possible with differential experimental data contained in the CSISRS (or EXFOR) database (NRDC-Network, 2017), while simultaneously showing general agreement with critical benchmark measured data. Evaluators did not make any changes to $\bar{\nu}$ between the previous evaluation (ENDF/B-VII.0) and the current evaluation (ENDF/B-VII.1) (Chadwick, 2006). Therefore, the evaluation process of the ENDF/B-VII.0 version will be described with regard to $\bar{\nu}$. For the ENDF/B-VII.0 evaluation, the experimental database from the ENDF/B-VI evaluation was used, with corrections to the normalization of the $\bar{\nu}$ nuclear data. This resulted in evaluations that match well with the corrected experimental database for ^{235}U, ^{238}U and ^{239}Pu. Appreciable deviation from experimental data occurs in the energy range below 1.5 MeV for ^{239}Pu, and this is partially due to the desire to match JEZEBEL (a LANL fast critical benchmark experiment) results in particular (Chadwick, 2006).

One of the main parameters of interest that is used to validate ENDF is the effective multiplication factor k_{eff}, which is sensitive to $\bar{\nu}$ but not to the other moments of the $P(\nu)$ distribution. The effective multiplication factor is in general insensitive to changes in the correlated physics of fission and depends only on averages. This can be illustrated by examining the neutron transport equation, which consists of terms representing the loss of neutrons due to leakage out of the system, the loss of neutrons due to all interactions, the addition of neutrons due to in-scattering from another energy group, and the production of neutrons due to fission. Only the average quantity $\bar{\nu}(E)$ is required to calculate neutron transport and the effective multiplication factor of a system. However, by looking at the Hage-Cifarelli equation for the leakage multiplication of a system, in Eqs. (11) and (12), which are presented and explained later in this work, it is clear that other moments of the multiplicity distribution (ν_{s2}, ν_{I2}) are also important.

The average number of neutrons released per fission is a specific measured observable of some of the differential measurements of fission product yields, masses, and fission neutron energy spectra contained in EXFOR. Because of the contribution of neutrons from

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