Multiplicity Analysis During the Photon Interrogation of Fissionable Material

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The identification of fissile material in the presence of benign material is of great concern to the United States and the world. The principal difference between fissile and benign materials is the multiplicative behavior of the fissile sample. Consider the interrogation of a sample with high-energy photons in the presence of a detection system capable of accurately recording the emissions. During the interrogation of a benign material, such as lead, the reaction emissions would largely be single-particle in nature, whereas the interrogation of a fissile material would induce multi-particle fission reactions. These fission neutrons could cause further fissions in the sample. The result would be many particles arriving at the detectors within a given time window; a grouping of particles arriving at the detector together is generally referred to as a "multiplet." The distribution of these multiplets could hold a great deal of information regarding the nature of the interrogated sample. Additionally, the assumptions made regarding the number and distributions of neutrons generated in photon-induced fission will be investigated. The current version of the MCNP-PoliMi detection post-processor has the capability of computing the multiplicity distribution based on the information in the collision output file. However, the Monte Carlo method treats each history individually; this results in a drastic under-estimation of the multiplicity distribution. In order to accurately model the multiplicity distribution, the width of the interrogation pulse must be taken into account. This dictates how many photons arrive at the target simultaneously. In order to model the pulse width correctly, a post-processing algorithm has been developed which operates on the MCNP-PoliMi collision output file. The purpose of this subroutine is to assemble the interactions into *groups* corresponding to the number of interactions that would occur during a given pulse. After the output file has been reordered, a multiplicity analysis may be performed. This capability will enable the simulation of a large number of materials and detector geometries. Analysis of this information may determine the feasibility of using multiplicity distributions as an identification tool for special nuclear material.

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