Sean Martinson, Texas A&M University
Title: Validation of MCNP through Isotopic Analysis of LEU UO2

Abstract
Monte Carlo (MC) neutronic codes are tools that may aid in the understanding and simulation of the harsh and complex environment within a nuclear reactor. MC neutronic codes can model neutron transport calculations through stochastic random sampling. One commonly used MC code is Monte Carlo N-Particle (MCNP), developed by Los Alamos National Laboratory. MCNP can be used to model simple geometries of an infinite slab reactor to an intricate fast breeder reactor. The validity of the models produced by MCNP may only be as strong as the accuracy of the code itself. There are many benchmark studies that look at MCNP's ability to simulate nuclear criticality, but there are fewer studies that examine isotope production. By reviewing these studies, it was found that MCNP can accurately predict the production of isotopes including actinides like 239Pu and 240Pu, and fission products such as 149Sm, 154Eu, 136 Ba, and 137Cs, to name a few. Conversely, MCNP has been shown to poorly estimate 125Sb and isotopes of curium (e.g. 243Cm, 244Cm, 245Cm, etc.). The experimental work for this study will start with an MCNP simulation of a sample of LEU UO2 (~100 mg) being irradiated for one month in the University of Missouri Research Reactor (MURR). The irradiated sample will then be cooled for an additional month. After the simulation is complete, the experiment will be carried with a real sample and the MURR. Once the sample is returned, a chemical dissolution and separation will be done. Finally, the sample will be analyzed by ICP-MS and select isotopes will be compared to the MCNP predictions. The results of this study will provide evidence for the accuracy of MCNP and its utility for nuclear forensics applications involving LEU UO2.