Nuclear Data Error Propagation in Fusion Benchmark Calculations

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INTRODUCTION

The SINBAD Database [1] is a collection of experimental benchmarks that are used for a variety of shielding applications including fission and fusion reactors and accelerators. The purpose of the current study is to calculate sensitivity coefficients and perform uncertainty propagation in fixed-source problems. For this purpose, benchmark problems were selected from the fusion shielding benchmarks of the SINBAD database. For the selected benchmarks, sensitivity parameters have been calculated and used in uncertainty propagation. Until recently, uncertainty propagation had been primarily focused on eigenvalue problems [2,3,4] for fission-based systems.

A prototype sensitivity and uncertainty analysis sequence has been developed at the Oak Ridge National Laboratory (ORNL) to allow sensitivity calculations and uncertainty propagation for fixed source problems. Design applications for fusion systems require uncertainty propagation and sensitivity analysis to determine optimum system parameters such as fusion reactor blanket, tritium breeding ratio, etc. The ORNL fixed-source sensitivity analysis tool was integrated into the TSUNAMI-1D sequence of the SCALE code system.[5] TSUNAMI-1D employs the Adjoint-Based Perturbation Theory (ABPT) for determining sensitivity coefficients for one-dimensional (1D) multi-group transport calculations.

GENERATION OF SENSITIVITY COEFFICIENTS WITH ONE-DIMENSIONAL FIXED-SOURCE CALCULATIONS

From the SINBAD database a total of five one-dimensional spherical geometry benchmarks were selected. The shielding materials contained in each of these five benchmarks are, respectively, Al, Ni, Fe, W, and stainless steel. The MCNP code [6] with ENDF/B-VII.0 cross-section data was used as the reference solution for the 1D benchmark models. The TSUNAMI-1D sequence of the SCALE code system was used to simulate the neutron population of the benchmark systems using the 1D discrete ordinates (S_n) transport module XSDRNPM to calculate the forward and adjoint fluxes and the SAMS module to generate the sensitivity and uncertainty data. XSDRNPM contains several features that make it suitable for use in nuclear data uncertainty propagation for fixed-source shielding benchmark problems. The first major advantage of XSDRNPM is its capability of computing energy-dependent forward and adjoint fluxes that are used for quantifying reaction rates. The other major benefit of XSDRNPM is its ability to handle distributed neutron fixed-source calculations required to accurately model benchmarks. Two ENDF/B VII.0 multi-group neutron cross-section libraries of the SCALE system, namely, the 238-group, and the Vitamin B7 199-group were used in the calculations. The uncertainty analysis is based on the 44-group SCALE covariance library.

The sensitivity coefficient of a given response to a specific nuclear data parameter is given by Eq. (1),

$$S = \frac{\sigma}{R_i}\frac{dR_i}{d\sigma}, \tag{1}$$

where σ is the nuclear data parameter and R_i is the response function. The sensitivity coefficient of a given response to the fixed source is given by Eq. (2),

$$S = \frac{q_n}{R_i}\frac{dR_i}{dq_n}, \tag{2}$$

where q_n is the fixed source and R_i is again the response function of interest. The derivatives for Eq. (1) and Eq. (2) are calculated using the ABPT method, which requires a single forward calculation, along with one adjoint calculation for each of the responses.

Three figures of merit (FOMs) are of importance to the fusion shielding benchmark problems, namely neutron activation analysis, kerma factors, and dose calculations. Sensitivity calculations were carried out for these quantities. As an example, the numerical results for the sensitivity of neutron activation to the Al inelastic scattering cross sections as a function of energy are displayed in Fig. 1. As can be seen, the response (neutron activation) is very sensitive to the inelastic scattering cross section. The inelastic scattering dominated the scattering for all of the shielding materials that were used

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in the benchmark problems. Furthermore, the uncertainty of neutron activation due to the inelastic scattering cross section of Al was calculated to be 2.24×10^{-4} %. The reason that the uncertainty in the Al cross-section is rather low is due to the fact that the Al cross-section has been well studied over the years. The overall uncertainty in neutron activation due to all covariance data was 4.78%. Neutron interaction with nitrogen in the air contributes to a large source of uncertainty for the current studied benchmark, 2.12%. This study indicates that a better understanding and improvement of the neutron cross-section for nitrogen is needed to reduce the overall uncertainty in the response. .

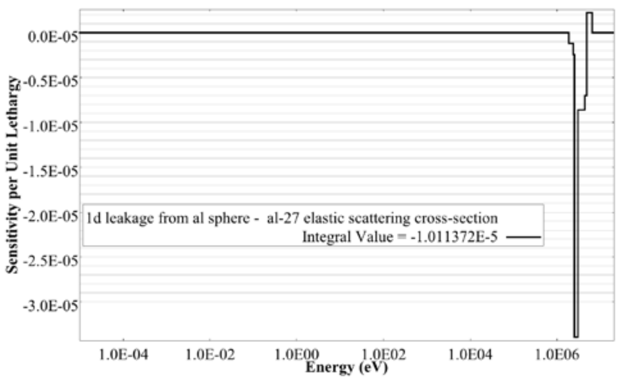

Fig. 1. Sensitivity coefficient of Al inelastic scattering cross section versus source neutron energy.

CONCLUSIONS

In summary, this work demonstrates the feasibility of calculating sensitivity coefficients and performing uncertainty propagation on a fixed-source problem. The impact of nuclear data uncertainty in a fusion benchmark calculation has been demonstrated using ORNL TSUNAMI-1D capability. In addition, the procedure presented here can be used to help verify the validity of data uncertainty in benchmark calculations.

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