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New Monte Carlo-based method to evaluate fission fraction uncertainties for the reactor antineutrino experiment

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Abstract

Uncertainties regarding fission fractions are essential in understanding antineutrino flux predictions in reactor antineutrino experiments. A new Monte Carlo-based method to evaluate the covariance coefficients between isotopes is proposed. The covariance coefficients are found to vary with reactor burnup and may change from positive to negative because of balance effects in fissioning. For example, between ^{235}U and ^{239}Pu, the covariance coefficient changes from 0.15 to −0.13. Using the equation relating fission fraction and atomic density, consistent uncertainties in the fission fraction and covariance matrix were obtained. The antineutrino flux uncertainty is 0.55%, which does not vary with reactor burnup. The new value is about 8.3% smaller.

Keywords: Monte Carlo-based method; Fission fraction uncertainty; Reactor antineutrino experiment

1. Introduction

Reactor antineutrinos are used to study neutrino oscillations, searches for signatures of non-standard neutrino interactions, and monitoring reactions to safeguard operations. The antineutrino flux is a main source of uncertainty in reactor neutrino experiments. The antineutrino flux

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and the spectrum of $\bar{\nu}_e$ from a reactor on any given day t can be predicted using the equation for the total antineutrino flux $S(t)$ from the reactor core,

$$S(t) = \int_{E_\nu} \frac{W_{th}(t)}{\sum_i f_i(t) E_i} \sum_i f_i(t) S_i(E_\nu) dE_\nu \tag{1}$$

where W_{th} is the thermal power of the reactor, and f_i, E_i, and $S_i(E_\nu)$ are the fission fraction, energy release, and antineutrino spectrum of isotope i, respectively; ($i = {}^{235}$U, ^{238}U, ^{239}Pu and ^{241}Pu). The uncertainty associated with the antineutrino flux by the fission fraction can be calculated from

$$\frac{\delta S}{S} = \frac{1}{S} \sqrt{\sum_{i,j} \frac{\partial S}{\partial f_i} \frac{\partial S}{\partial f_j} \cdot \delta f_i \delta f_j \rho_{i,j}}, \tag{2}$$

where δf_i is the uncertainty associated with the fission fraction f_i, and $\rho_{i,j}$ are the correlation coefficients between isotopes given by

$$\rho_{i,j} = \frac{1}{N-1} \sum_{k=1}^{N} \frac{(f_i - \bar{f}_i)}{\sigma_{f_i}} \frac{(f_j - \bar{f}_j)}{\sigma_{f_j}} \tag{3}$$

where N is the total sample number, and $\bar{f}_i$ and σ_{f_i} are the average and standard deviation of f_i, respectively.

During the power cycle of a nuclear reactor, the composition of the fuel changes as Pu isotopes are bred and U isotopes are depleted. At the end of the power cycle, some fraction of the fuel is replaced; for example, for the Daya Bay reactor, one third of the fuel is replaced. Fresh fuel always stays in the reactor for three cycles in order to generate more power. Understanding the detailed time-dependence of the fuel content is of practical interest for reactor operators and designers particularly in regard to safety considerations. To obtain these details, reactor simulation codes were developed such as DRAGON [2], Reactor Monte Carlo code (RMC) [3], CASMO, and SCALE [4]. The verification and validation of these simulation codes were conducted by comparing the results of isotopic concentrations with experimental results. The coefficients of fission fractions between isotopes [1] were approximately studied using 159 comparisons of fuel element samples taken from ten pressurized water reactors (PWRs) and boiling water reactors (BWRs), modeled by a variety of core simulation codes because these isotopic concentration comparisons only give indirect information on the uncertainty in the number of fission fractions. The correlation coefficients was also studied using fission fractions of each isotope directly [5]. However, for the next generation reactor antineutrino experiment (JUNO), which aims to perform high-precision neutrino oscillation measurements, more detailed information is needed about the uncertainties associated with fission fractions, such as specifying the relationship between the uncertainty in the flux prediction caused by fission fraction and reactor burnup. Previous studies give only average fission fraction uncertainty for reactor burnup, but not details concerning reactor burnup. In this study, this is addressed using a new Monte Carlo (MC)-based method.

In Section 2, the MC-based method of evaluating the correlation coefficient between different isotopes was introduced. The parameters, which are needed in the method, such as atomic density and one-group microscopic cross sections as a function of reactor burnup, are calculated using RMC. The model for this calculation is discussed in Section 3. The covariance matrix results are discussed in Section 4, and the relation between mass inventory and fission rates are discussed