Investigation of the approach used in the unresolved resonance region

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The typical practice used in Monte Carlo neutron transport codes in the unresolved resonance region (URR) is to take advantage of the probability table (PT) approach. Cross sections are sampled from PTs and used as needed, along with generated average cross sections. The PTs are generated based on cross-section calculations performed using the single-level Breit–Wigner approximation. Although the approach used in the URR seems plausible, a detailed examination was needed to understand the benchmark results obtained using specific tests.

The strategy used in this effort was to build a critical benchmark (computational benchmark) with values of energy corresponding to average lethargy of neutrons causing fission (EALF) in the region where the URR contribution is predominant. For this purpose, a critical ^{238}U-based benchmark with an EALF of 1.3 keV was built. The resolved resonance region (RRR) of ^{238}U extends up to 20 keV. The method of evaluating ^{238}U was modified: above 100 eV, the URR approach was used instead of the RRR approach. Recall that from 100 eV to 20 keV, the cross-section values in the original ^{238}U evaluation are based on true cross sections. Additionally, the original evaluation was performed using the Reich–Moore formalism. Results based on the original ^{238}U evaluation and those in which modified RRR were converted into URR were investigated. Calculations were performed using the MCNP6 and TRIPOLI-4 codes, based on the Evaluated Nuclear Data File (ENDF)/B-VII.1 and Joint Evaluated Fission and Fusion File (JEFF) 3.3 cross-section libraries, respectively.