Nuclear Data Evaluation for Reactor Applications

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Introduction

In past years, nuclear analysts had to rely on a limited amount of nuclear data for reactor design. In that time, the need for nuclear data was driven by the thermal and fast reactor programs. In a thermal reactor, the fissions of importance all take place below 4 eV. Because of this, evaluations for thermal applications emphasized this region. In a typical fast reactor, however, the most important fission range shifts upwards to the 10's to 100's of keV region, and this led to evaluations that emphasized this range. Nuclear criticality situations involve an energy spectrum that peaks in the 10's to 100's eV range, and need critical attention to the cross sections in this range. Since all of these systems produce fission neutrons at high energies-500 keV to a few MeV-attention has been given to this energy range. As noted, the region most neglected is the epithermal region; however, calculational experiences suggest many nuclides need improvements in all energy regions.

Description of the actual work

In the early 1980's, a data measurement and evaluation program was initiated at the Oak Ridge National Laboratory (ORNL). Measurement activities were carried out at the Oak Ridge Electron Linear Accelerator (ORELA) and data evaluations were performed utilizing the computer code SAMMY.[1] The intent of the program was to generate evaluated data for the majority of important nuclides such as ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, and for structural materials, such as iron, nickel, and to ultimately replace the corresponding evaluations in the ENDF/B-V and generate a new evaluated nuclear data library, the ENDF/B-VI library because concerns regarding the earlier evaluations. For example, the ENDF/B-V representation of the ^{235}U cross sections in the resolved resonance region is unsatisfactory; the evaluation is based on the Single-Level Breit-Wigner formalism coupled with “smooth files”(file 3 representation), with no information provided for the resonance spin, and with the resonance representation extending only up to 82 eV. Other nuclide evaluations in ENDF/B-V follow the same characteristic of the ^{235}U evaluation. The new resonance-region evaluation for ^{235}U was done using the Reich Moore resonance formalism to exactly treat the interference effects in the fission channels. Indeed, the Reich Moore formalism was adopted as the cross section representation for ENDF/B-VI for most of the actinides and structural materials. After several releases of the ENDF/B-VI library (presently release 5), it appears that the ^{235}U evaluation performs satisfactorily for system with energy spectrum in the thermal, intermediate, and fast energy region. The ENDF/B-VI ^{235}U evaluation has been included in the Joint Evaluated File (JEF) and with minor changes, in the Japanese Nuclear Data Library (JENDL).

Data evaluations at ORNL are now done include both differential and integral data. This approach provides cross section data that calculate integral benchmarks more accurately than in the past. Antecedent evaluations in previous releases of ENDF/B-VI did not use this procedure.

Although the ENDF/B-VI evaluations are superior to the ENDF/B-V evaluations, some problems still remain: Benchmark analysis of mixed-oxide and UO_2 assemblies in a boiling water reactor with fissile plutonium has been done to address the issue of disposition of weapons-grade plutonium; the study suggested that the cross sections for ^{239}Pu in the ENDF/B-VI library need to be revised to improve the computed and the measured ratio of the isotopic composition for different burnups.[2] A combination of differential and integral data in the re-evaluation of ^{239}Pu would definitely improve the cross section data for this isotope. A second concern relates to a longstanding problem with the ^{238}U cross sections for calculating lattice benchmarks.[3] Although at ORNL we have not made a detailed study on this subject, it appears that ^{238}U cross sections in the unresolved energy region needs to be revised.

Results

Criticality safety for treating nuclear material outside reactors has emerged as an area which requires accurate nuclear data. Unlike thermal and fast systems, the neutron energy spectrum for criticality safety applications peaks in the intermediate energy region. ORNL is presently performing several cross section measurements to improve the cross section representation in the intermediate energy region. Recently a Reich-Moore evaluation of the ^{233}U cross sections has been performed for ENDF/B-VI, as concerns had originated from the prediction of critical mass calculations of ^{233}U using existing cross section libraries. A comparison of the ^{233}U fission cross section measured at ORNL and the existing ENDF/B-V evaluation is shown in Fig. 1. The solid line shows the average cross section as calculated with ENDF/B-V and the circles joined by a solid line shows the ORNL measured cross section. Clearly, ENDF/B-V does not provide accurate details in the cross section. Thermal benchmark calculations using ORNL and ENDF/B-V evaluations are shown in Table 1; these benchmarks are critical spheres of ^{233}U moderate by water. Results using ^{233}U ORNL evaluations show improvements versus ENDF/B-V results. Calculations were done using the KENOV.a code with 199-group structure of the VITAMIN-B6 library.

In conclusion, improvement in reactor physics calculations require accurate evaluated nuclear data. This can be achieved only by performing high-resolution data measurements and using rigorous formalisms to evaluate nuclear data.

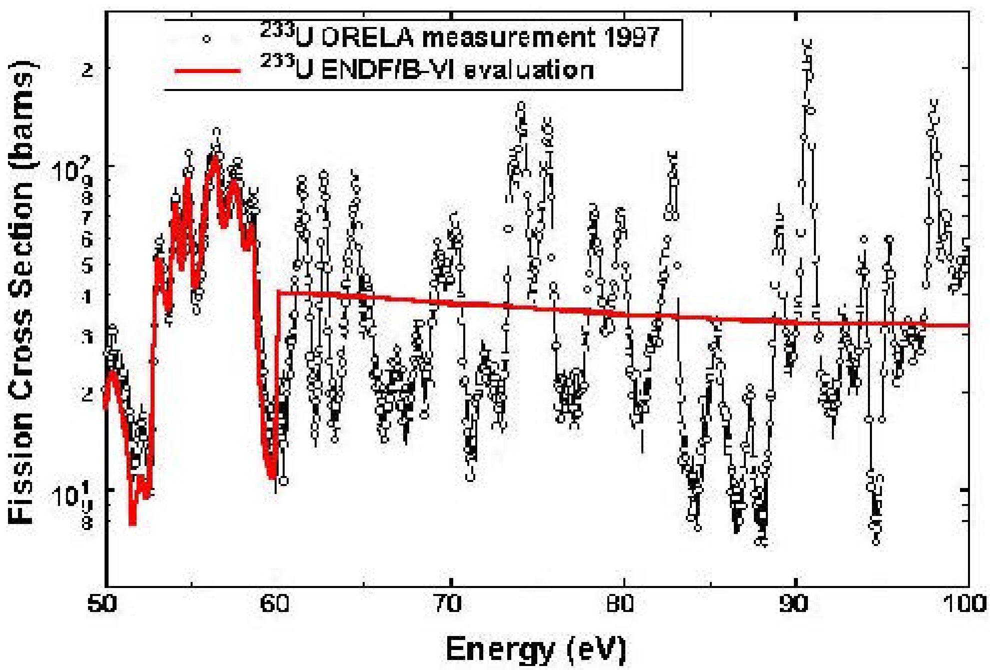

Figure 1 Comparison of ORNL measured ^{233}U fission cross sections with the existing ENDF/B-V evaluation.

Table 1. Comparison of the ^{233}U ORNL evaluation and ENDF/B-V evaluation with thermal energy benchmarks using KENOV.a code with the 199-group structure of the VITAMIN-B6 library.

Benchmark	ENDF/B-V	ORNL
ORNL-5	0.9964 +/- 0.0008	1.0006 +/- 0.0009
ORNL-6	0.9962 +/- 0.0009	0.9997 +/- 0.0008
ORNL-7	0.9948 +/- 0.0008	0.9996 +/- 0.0008
ORNL-8	0.9963 +/- 0.0008	1.0000 +/- 0.0009
ORNL-9	0.9950 +/- 0.0008	0.9998 +/- 0.0007
ORNL-11	0.9951 +/- 0.0005	0.9987 +/- 0.0006

Reference:

1. N. M. Larson, *Updated Users' Guide for SAMMY: Multilevel R-matrix Fits to Neutron Data Using Bayes' Equations*, ORNL/TM-9179, Martin Marietta Energy Systems, Inc., Oak Ridge National Laboratory (August 1984) see also ORNL/TM-9179/R1 (July 85), /R2 (June 1989), /R3 (September 1996), /R4 (December 1998), and /R5 (October 2000).

2. S. E. Fisher and F. C. Difilippo, "Neutronic Benchmark for the Quad Cities-1 (Cycle 2) Mixed Oxide Assembly Irradiation," ORNL/TM-13567, April 1998.

3. R. D. Mosteller, "Helios Calculations for UO_2 Lattice Benchmarks," International Conference on the Physics of Nuclear Science and Technology, Long Island, New York, October 5-8, 1998.