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Benchmark Data Testing for Thermal-Reactor Applications*

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The purpose of the thermal reactor data testing effort is to determine the adequacy of basic cross section data in thermal reactor applications, and particularly for beginning of life criticality calculations.

This data testing effort has:

- Pointed up inadequacies in Evaluated Nuclear Data (ENDF/B) sets, and thus has stimulated new differential data measurements and improved evaluations.
- Provided standard calculated results against which approximate or newly developed accurate methods can be tested.
- Identified inadequacies in Integral Experiments.
- Identfied problems in specific calculational methods.

In preparation of ENDF/B-V a great deal of testing has been done with ENDF/B-IV as well as preliminary versions of ENDF/B-V. This data testing, covering the major fissile and fertile materials, highlighted a number of discrepancies, and contributed to a number of data improvements by the nuclear evaluators which have significantly upgraded the performance of ENDF/B-V for thermal reactor analysis.

Data testing has centered around a series of benchmark assemblies for the principal fissile and fertile materials. These include homogenous aqueous assemblies and simple rod lattices. The Cross Section Evaluation Working Group (CSEWG) a cooperative effort of over 20 laboratories, has specified a number of thermal benchmarks which are either H_2O or D_2O moderated and most are critical assemblies. Detailed Benchmark specifications are provided in ENDF-202.[1]

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In the initial CSEWG testing of ENDF/B-IV effort was concentrated on the homogenous assemblies and the simple lattices of slightly enriched uranium rods in a moderator of H_2O or D_2O. Integral transport theory with special resonance treatments, and Monte Carlo methods were used. Results of these efforts are collected in an ENDF/B-IV Data Testing Report.[2]

The ORNL aqueous U235-H_2O assemblies covered a wide range of neutron leakage through variations of the H/U235 fraction. The calculated k_{eff} compared to experiment for these assemblies significantly increased with leakage. This was attributed to (1) a leakage under prediction caused by the soft fission spectrum in ENDF/B-IV, and (2) the high value of η for U235 in ENDF/B-IV which is emphasized in the harder flux spectrum of these high leakage assemblies. Evidence such as this strongly suggested to the evaluator a need for a harder fission spectrum for U235 which led to the adoption for ENDF/B-V of a prompt Watt spectrum with $\bar{E}$ = 2.03 MeV. (The ENDF/B-IV spectrum was Maxwellian with $\bar{E}$ = 1.985MeV.) The analysis also suggested the need for an increase of thermal $\bar{\nu}$ for U235. A 0.74% increase in $\bar{\nu}$ for U235 was adopted in ENDF/B-V compared to ENDF/B-IV.

The TRX assemblies have been extensively analyzed both from a data and methods point of view. The chief problem has been to properly calculate shielded U238 resonance capture. The strong correlation between calculated k_{eff} and the epithermal to thermal ratio of U238 capture (ρ^{28}) for these assemblies have been the chief source of discrepancy. Since the shielded resonance integral is only about 15 barns, unshielded components of the capture, which are small compared to the dilute capture integral become very important in these lattices. The calculated values of ρ^{28} decreased approximately 2% from ENDF/B-IV to V, due mostly to the impact of new measurements[3] reducing the capture width of the first 3 s-wave levels of U238 and increasing the neutron widths at 20.9 eV and 36.8 eV.

The calculation is now approximately 2% above experiment on the average and is consistent with the overall uncertainty assignments. The effective multiplication factor increased from 0.987 (ENDF/B-IV average) to k_{eff} = 0.997 with ENDF/B-V, due primarily to the 0.74% increase of the U235 thermal $\bar{\nu}$ normalization.

The aqueous plutonium (nitrate) -H_2O assemblies are mostly unreflected homogeneous spheres modeled as a sphere in a void. These assembles were analyzed with ENDF/B-IV by a number of laboratories. Average k_{eff} results are high (k_{eff} = 1.011), and show a considerable amount of scatter. The overall picture using ENDF/B-V data is unchanged from Version IV. The large scatter of k_{eff} (2.7%) shows no convincing trends and is probably attributable to uncertainties in plutonium content. In any case, sensitivities are sufficiently small that it is difficult to explain the high k_{eff} on the basis of obvious data deficiencies. There remains a need for several highly accurate benchmark critical experiments and careful studies to identify the source of high k_{eff} values

An ENDF/B-V Data Testing Report has been prepared,[4] and it is instructive to complete our comments about the thermal benchmark coverage with the information in Table 1 from that report which summarizes the current status of the experiments.

T ble 1

Benchmarks Characteristics

System Type	Benchmark Coverage
U235-H_2O	Good. Small assemblies should be added to benchmark set.
U235-U238-H_2O	Good.
U235-U238-D_2O	Poor. MIT-4, 5, 6 are old and exponentials. LTRIIA is complex and has some inconsistencies with ρ^{28}. ZEEP is being added.
U233-H_2O	Good. Full range of ORNL and small assemblies should be added to benchmark set.
U22-Th-H_2O	Good. BNL's are exponentials. Would be better to have simple critical lattices.
Pu-H_2O	Poor. Need some well determined experiments.
UO_2-PuO_2-H_2O	Fair. Could use additional experiments to resolve inconsistencies and provide lattice parameters.

References

1. H. Alter et al., "Cross Section Evaluation Working Group Benchmark Specifications," BNL-19302, ENDF-202 (1974).

2. E.M. Bohn et al., "Benchmark Testing of ENDF/B-IV," BNL-NCS-21118, ENDF-230 (1976).

3. G. de Saussure et al., "Evaluation of the ^{238}V Neutron Cross Sections for Incident Neutron Energies up to 4 keV," ORNL/TM-6152, ENDF-257 (1978).

4. CSEWG Data Testing and Applications Committee, "Benchmark Testing of ENDF/B-V," ENDF-311, to be published (1981).