DATA TESTING OF ENDF/B-VI

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ABSTRACT

A number of the fast reactor and thermal reactor benchmarks have been analyzed using nuclear data from ENDF/B-VI Release 2. Data were prepared with the NJOY nuclear data processing system in MATXS and ACE formats. Transport calculations were preformed with ONEDANT and TWODANT using transport tables prepared by the TRANSX code and with the MCNP Monte Carlo code.

I. INTRODUCTION

The Evaluated Nuclear Data Files (ENDF) have been very successful over the years in providing a source of standardized and reliable nuclear data for a wide variety of applications. The system is controlled by the Cross Section Evaluation Working Group (CSEWG), a committee with representatives from the national laboratories, industry, universities, and funding agencies. The National Nuclear Data Center (NNDC) at the Brookhaven National Laboratory maintains and distributes the ENDF data libraries. The data libraries have been issued in "versions" such as ENDF/B-IV, ENDF/B-V, and the new ENDF/B-VI.

A key component of the CSEWG program for releasing a new version is data testing, which is important for finding errors and for giving the users an initial indication of how well the new version would be expected to work for their applications. However, this process is very difficult, and the results are often ambiguous. Because of the difficulty, very simple models have been specified for the benchmarks. This makes it easier for a wide variety of laboratories to compute the benchmarks. This simplicity is also the source of the ambiguity. In many cases, model corrections were computed using previous data libraries and older methods. If a new calculation using ENDF/B-VI gives worse answer than ENDF/B-V, it may not really prove that the new version doesn't perform better than the previous one! In addition, experience in attempting to compute some benchmarks over the years suggests that the experiments were characterized incorrectly. No improvement in the data would be expected to improve the results for these assemblies.

Because of these kinds of problems, data testing results like the ones reported here must be used cautiously. Clearly the *differences* between ENDF/B-V results and ENDF/B-VI results are significant. If an organization knows how its own calculations using ENDF/B-V perform for its own applications, it can get a pretty good idea of how ENDF/B-VI will perform for the same applications. In addition, some benchmarks are easier to compute reliably than others. For example, the simple bare and reflected spherical assemblies usually give reliable information on the cross sections. For other assemblies, more complete methods and very detailed models have sometimes been used to generate corrections to the simple models.

Even with these limitations, data testing is an important part of the documentation for each new version of the ENDF/B files, and we have carried out calculations for some of the most important fast reactor and thermal reactor CSEWG benchmarks[1] using ENDF/B-VI through Release 2 (June 1993).

II. METHODS

The ENDF/B-VI data "tapes" were processed using the NJOY nuclear data processing system.[2] The new library contains a number of new features,[3] such as Reich-Moore resonance parameters and energy-angle distributions in File 6. All of the new features needed for the data testing calculations are handled by version 91.91 of NJOY. The final results were converted into MATXS format using several different combinations of group structure, weight function, self-shielding, thermal upscatter, and so on. The libraries called MATXS10, 11, 12, 13, and 14 use neutron structures with 30, 80, 69, 187, and 175 groups, respectively. MATXS 12 and 13 are suitable for thermal problems, and all but MATXS10 are suitable for problems where self-shielding is important. In order to get reliable measurements of the changes between the older ENDF libraries and ENDF/B-VI, new libraries were prepared for both ENDF/B-IV and ENDF/B-V using modern methods. MATXS1 and MATXS5 are 30-group sets for version IV and V, respectively. MATXS3 and MATXS7 are 69 group sets for the two versions. MATXS6 is an 80-group set for ENDF/B-V. We also prepared libraries in ACE format for the MCNP[4] Monte Carlo code for the key materials of ENDF/B-IV, -V, and -VI. Some of these libraries are already available from the Radiation Shielding Information Center (RSIC) at Oak Ridge.

MATXS libraries are used with the TRANSX code[5] to prepare transport tables for the actual benchmark calculations. Most of the results reported here were computed using the Los Alamos discrete-ordinates diffusion-accelerated transport codes ONEDANT[6] and TWODANT.[7] In these cases, TRANSX was used to generate macroscopic cross sections, and the mixing capabilities of the transport code were not used. This path is required for self-shielded cross sections and is convenient for the other cases. Another advantage of this path is that mixture and region-dependent fission spectra can be used. In some cases, two passes were made through TRANSX and the transport code. The first pass prepared and used a fission source computed using the default weight function for the library. The flux from this calculation was then used in the second TRANSX run to prepare flux, mixture, and region-dependent fission spectra for the second transport calculation. All self-shielding effects were included using simple narrow-resonance equivalence theory.

III. THE ASSEMBLIES

Over the years, the Los Alamos National Laboratory has built, operated, and analyzed a number of small, fast critical assemblies. These assemblies are characterized by simple compositions and geometries that make them amenable to accurate calculation. Therefore, computing these assemblies can be truly called "data testing." Because of their small size, they only test the cross sections at fairly high energies. GODIVA is a sphere of enriched uranium, JEZEBEL is a sphere of clean plutonium, JEZEBEL-23 is a sphere of ^{233}U, and JEZEBEL-PU is a sphere of "dirty" plutonium. There are also corresponding criticals with ^{238}U reflectors around the cores: FLATTOP-25 has a ^{235}U core, FLATTOP-23 has a ^{233}U core, and FLATTOP-PU has a core of clean plutonium. Finally, BIGTEN is a larger assembly consisting of interleaved plates of enriched and depleted uranium. It produces a somewhat softer neutron flux than the other assemblies. Calculations have been carried out for these assemblies using a variety of group structures, scattering orders, spatial grids, and so on. Either 30 or 80 groups gives reasonable results, and P_3/S_8 transport is usually sufficient. The exception is BIGTEN, where 30 groups is not enough, and where self-shielded cross sections are needed. Table 1 summarizes the k_{eff} results.

Table 1. Computed k_{eff} Values
for CSEWG Benchmarks

Assembly	ENDF/B-V	ENDF/B-VI
GODIVA	.9990	.9983
JEZEBEL	.9982	.9989
JEZEBEL-23	.9940	.9940
JEZEBEL-PU	.9983	.9981
FLATTOP-25	1.0054	1.0030
FLATTOP-23	1.0041	1.0041
FLATTOP-PU	1.0050	1.0055
BIGTEN	1.0093	1.0105
ZPR-6/6A	.9894	1.0063
ZPR-6/6A	.9997	1.0070
ORNL-1	1.0003	.9969
ORNL-2	1.0000	.9967
ORNL-10	1.0000	.9972
L7	1.0069	1.0050
L8	1.0082	1.0044
L9	1.0060	1.0025
PNL-1	1.0166	1.0087
PNL-2	1.0092	1.0037
PNL-3	.9978	.9904
PNL-4	1.0049	.9971
TRX-1	.9881	.9869
TRX-2	.9912	.9891
BAPL-1	.9970	.9949
BAPL-2	.9982	.9959
BAPL-3	1.0001	.9974

The Liquid-Metal Fast Breeder Reactor (LMFBR) criticals operated by the Argonne National Laboratory (ANL) test the ENDF cross sections at slightly lower energies. ZPR-6/6A was a uranium-fueled assembly, and ZPR-6/7 used plutonium fuel. ANL constructed simplified one-dimensional models for these assemblies back in the days of ENDF/B-III, and they provided correction factors to handle complexities like plate heterogeneity and streaming. It is not clear whether these corrections are still appropritate. These assembles require self-shielded cross sections, and 80 groups seems to be sufficient. The results for k_{eff} are in Table 1.

The lowest range of energies is tested by the thermal-reactor benchmarks. We have computed a number of homogeneous, spherical, aqueous, uranium assemblies (ORNL-1,ORNL-2,ORNL-10, L7, L8, and L9), aqueous plutonium assembles (PNL-1, PNL-2, PNL-3, and PNL-4), uranium metal rod lattices (TRX-1 and TRX-2), and uranium oxide rod lattices (BAPL-1, BAPL-2, and BAPL-3). The models for the spherical solution criticals are fairly simple and used 187 groups. The lattices were done using 69 groups and a two-step calculation. First, detailed fluxes inside each cell were obtained in an infinite lattice, and then the cells were homogenized and used for a full-core calculation using one-dimensional cylindrical geometry with a buckling height. The results are in Table 1.

IV. DISCUSSION

A review of the results for the Los Alamos criticals shows that the effects of going from ENDF/B-V to ENDF/B-VI are modest. In addition to the k_{eff} values summarized in Table 1, there are a large number of ratios, such as $\sigma_f(^{238}U)/\sigma_f(^{235}U)$, available. Some of the ratios are better, and some are worse, but there seems to be an increase in consistency between uranium and plutonium results. Also note that there is a consistent bias for k_{eff} to be higher for reflected assemblies than for the corresponding bare assembly. This bias shows up in ENDF/B-IV and -V also. The apparant high k value for BIGTEN is reduced when a two-dimensional calculation is used.

The results for the ANL LMFBR criticals appear to be high, but note that the uranium and plutonium results are more consistent. These assemblies are reflected with depleted uranium, which may help explain the high multiplication. The results for BIGTEN and the ZPR assemblies taken together suggest that ^{238}U absorption at high energies may be a little low.

The aqueous uranium assemblies have always shown a bias toward high k_{eff} for leaky assemblies (L7, L9), and this tendency persists for ENDF/B-VI. The multiplication for the very large spheres is now too low, and work is going on in CSEWG to correct both of these problems. The aqueous plutonium assemblies have always been a problem, and errors in the benchmark descriptions are suspected.

Our model for analyzing the lattice criticals produces results for k_{eff} that are lower by several tenths of a percent than those reported for previous CSEWG testing cycles.[8] This appears to be a consequence of the leakage treatment, because infinite-cell Monte Carlo calculations are in fair agreement with the simplified self-shielding results obtained using the multigroup methods. The primary source of the low predictions has persisted through versions IV, V, and VI; it is an overestimation of the epithermal ^{238}U captures. Repairing the under prediction of ^{235}U reactivity now found in ENDF/B-VI should help here also.

In summary, these results show some gains and some losses. The main worries are in the thermal range, and CSEWG is in the process of correcting the problems. We hope that this will make converting to ENDF/B-VI data attractive to the entire nuclear community.

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