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TITLE: Processing and Testing ENDF/B-VI With NJoy and Transx

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PROCESSING AND TESTING ENDF/B-VI WITH NJOY AND TRANSX

by

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ABSTRACT

The NJOY Nuclear Data Processing System has been used to process the Evaluated Nuclear Data File ENDF/B-VI, including all changes through Release 2, into multigroup cross-section libraries that can be used with many different neutron/photon transport codes and into ACE format for use with the MCNP continuous-energy Monte Carlo code. The ENDF-6 format contains a number of new features of interest to reactor physics, and many of the new evaluations take advantage of these features to provided improved data. The changes to the NJOY system required to process these new evaluations are described. The new cross-section libraries have been used to test the performance of ENDF/B-VI for a number of familiar benchmarks. The results are compared to ENDF/B-V results to give an idea of how the new libraries will perform for reactor physics calculations.

INTRODUCTION

The NJOY Nuclear Data Processing System[1] is widely used for preparing libraries of nuclear data for reactor physics calculations from evaluated data in the Evaluated Nuclear Data File (ENDF) format.[2] Recently, NJOY has been used to process the latest version of the US national file, ENDF/B-VI, including all changes through Release 2, into multigroup cross section libraries that can be used with many different neutron/photon transport codes and into ACE format for use with the MCNP continuous-energy Monte Carlo code.[3] The ENDF-6 format contains a number of new features. Evaluators have taken advantage of these new formats to produce new evaluations for many materials important to reactor physics, and some examples are discussed below. The changes to the NJOY code required to handle these new evaluations are also discussed below. In order to test both ENDF/B-VI and the NJOY processing code, we have used the new libraries to analyze a number of familiar benchmark assemblies from the Cross Section Evaluation Working Group (CSEWG) set.[4] The results are described below and compared to the corresponding results using ENDF/B-V data.

ENDF/B-VI FEATURES

The ENDF/B-VI files contain a number of new features of interest to reactor physics. Some examples include ^{235}U, ^{238}U, and ^{239}Pu with resonance ranges extended to higher energies to minimize problems with the unresolved range, and the isotopes of Fe, Ni, and Cr with new Reich-Moore resonance evaluations and energy-angle distributions for emitted neutrons and charged particles. These features improve calculations of transport through thick structures and give much better KERMA values.

Figure 1 shows the total cross section for ^{235}U in an energy region that was treated as unresolved for ENDF/B-V. There is a dramatic increase in the quality of the data. For ENDF/B-VI, the upper limit of the resolved resonance range has been changed from 82 eV to 2.25 keV for ^{235}U, from 4 keV to 10 keV for ^{238}U, and from 300 eV to 1 keV for ^{239}Pu. These changes should improve the representation of self-shielding by avoiding the use of unresolved-resonance methods in important energy ranges. In addition, multilevel Reich-Moore parameters are now provided, which give good line shapes and eliminate the need for the background corrections common with single-level representations.

Figure 2 compares the KERMA factors for ENDF/B-V.2 and ENDF/B-VI.2 for chromium. The new ENDF/B-VI evaluations are for separate isotopes, and the evaluators have provided explicit energy distributions for the recoil nuclei and charged particles emitted for each reaction. This makes it possible to compute heat production accurately without having to make dangerous assumptions about the reaction mechanisms during processing. For the ENDF/B-V evaluation, it was necessary to estimate the charged particle energy by subtracting the neutron energy and the energy in emitted photons from the available energy, which is a technique that requires extraordinary accuracy for the neutron and photon distributions given in the evaluation. Figure 3 compares the radiation damage parameters computed using ENDF/B-V.2 and ENDF/B-VI.2. The differences are smaller here, but the new numbers should be more reliable, because they are based on direct information provided by the evaluator, not on any kinematic models built into the processing code.

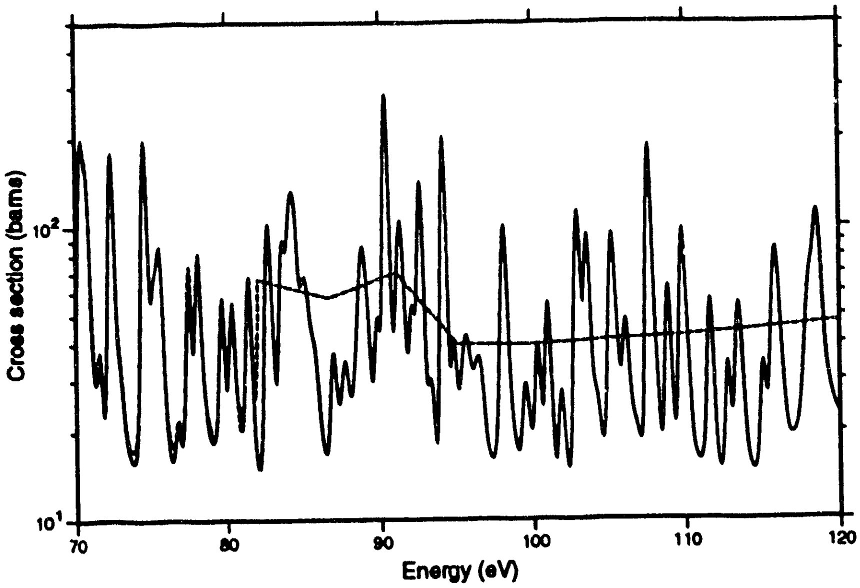

Figure 1: Total cross section for ^{235}U showing that part of the range that was given as unresolved in ENDF/B-V (dashed) is well resolved in ENDF/B-VI (solid).

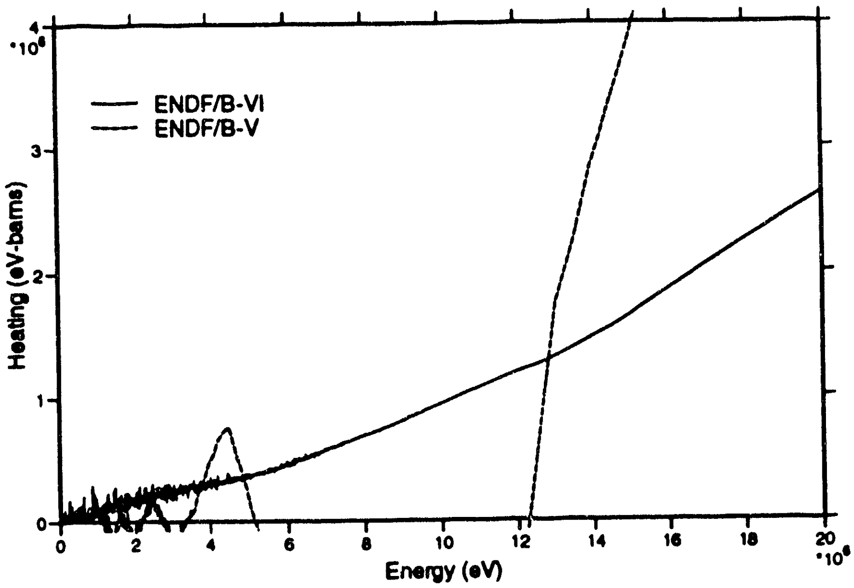

Figure 2: KERMA comparison for chromium.

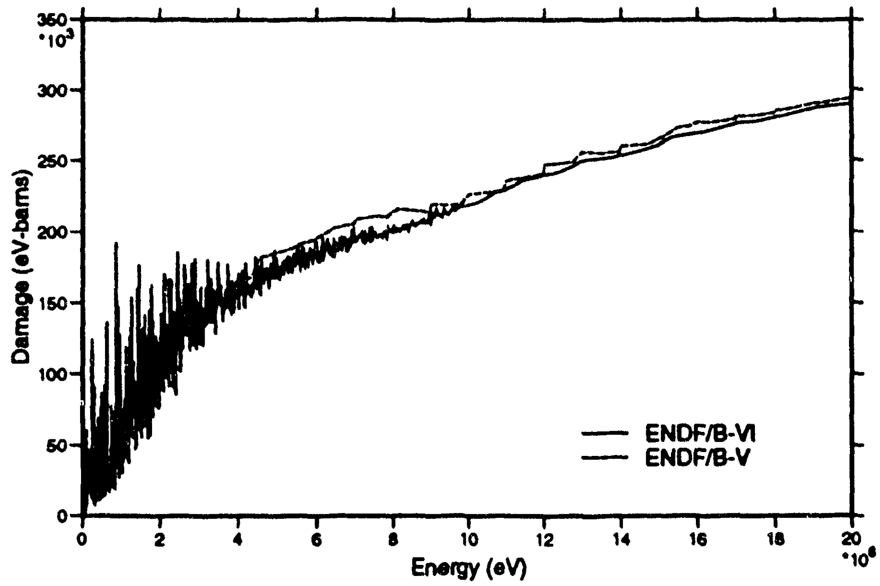

Figure 3: Damage comparison for chromium.

The new "File 6" format available in ENDF/B-VI allows for coupled energy-angle distributions for outgoing particles. Several different representations are provided, including Kalbach systematics in the CM (used by LANL for several actinides), Legendre distributions in the lab (used by ORNL for Fe, Ni, Cr, Cu, Pb, etc.), angle-energy distributions in the lab (used by LLNL for ^{9}Be), and phase-space distributions (used by LANL for ^{2}H). Energy-angle effects are most important for fusion and shielding problems; their impact on fission-reactor neutronics remains to be evaluated.

The ENDF-6 format has a number of improvements in the representation of thermal scattering (File 7). The initial set of evaluations for thermal moderators was produced by conversion from the original evaluations prepared for ENDF/B-III[5] and also used for ENDF/B-IV and ENDF/B-V. There were some errors in this conversion for coherent scattering in graphite, Be, and BeO. In addition, modern practice has led to using these evaluations for higher incident energies and larger energy transfers than was originally intended. A few new evaluations[6] have been prepared at Los Alamos for ENDF/B-VI.2 that provide for extended energy ranges, but they use the same physics models as the older evaluations. Figure 4 shows the increased energy-transfer range now possible for H in ZrH.

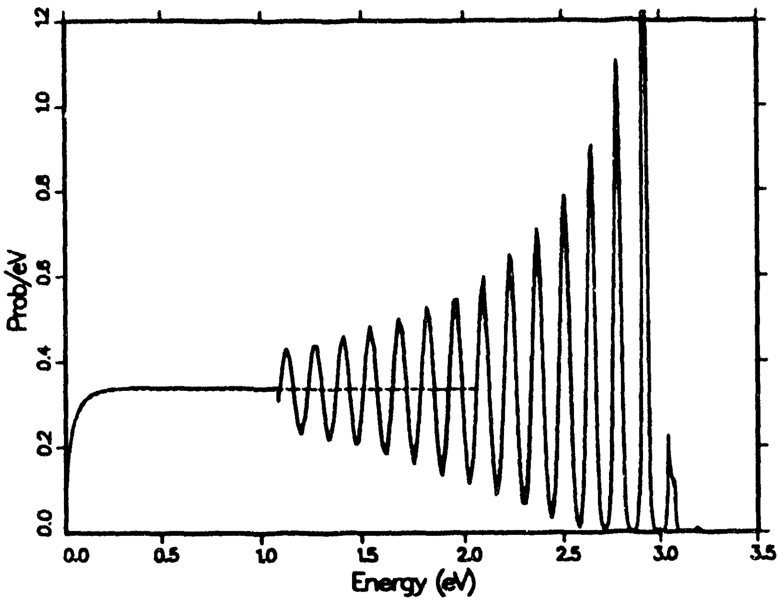

Figure 4: Neutron spectrum from H in ZrH for an energy of 3.059 eV. The solid curve is for the new evaluation, and the dashed curve is for the older evaluation.

NJOY DEVELOPMENT

Over the last several years, the NJOY code has been gradually upgraded to handle the new features of the ENDF-6 format. This effort has been aided by users at other laboratories in the US and around the world who have helped us to find and remove problems. Much of the coding to handle the new resonance options was provided by Charlie Dunford of the National Nuclear Data Center (NNDC), and the results were tested by an international effort including the IAEA Nuclear Data Section (Cullen, Ganesan), the NEA Data Bank (Sartori), and ORNL (Wright, Larson).

The development of the File 6 energy-angle capabilities was carried out at Los Alamos with some early involvement by Harm Gruppelaar (ECN Petten). Extensive changes were required in the HEATR module to compute heating and damage, in the GROUPR module to compute multigroup transfer matrices for emitted neutrons, charged particles, and photons, and in the ACER module for producing MCNP libraries. Most of the methods used have been described elsewhere.[7, 8] The most recent steps in this work consisted of getting NJOY and MCNP to work together properly for ENDF/B-VI data. This goal has been accomplished with the release of MCNP4A. The new File 6 capabilities were developed as a cooperation between Groups T-2 and X-6, with contributions from Bob Little, John Hendricks, and Bob Seamon. Legendre distributions as used by ORNL are currently approximated by converting them to Kalbach form for sampling.

PROCESSED LIBRARIES

In late 1993, the version NJOY 91.91 was used to process many of the materials of ENDF/B-VI.2 into several multigroup libraries in MATXS format. MATXS10 is a 30x12 (that is, 30 neutron groups and 12 photon groups) library that is compact and useful for high-energy problems where resonance effects are not important. MATXS11 is an 80x24 library with extensive self-shielding tables. It is useful for fast-reactor and fusion studies. MATXS12 is a 69x24 thermal reactor library with self-shielding and upscatter. MATXS13 is a 187x24 multipurpose library for fusion, fast-reactor, and thermal-reactor problems. MATXS14 is a 175x42 library for fusion neutronics studies. We have also prepared partial ENDF/B-V libraries using matching specifications for use in comparisons and for establishing bias factors against previous results; they are MATXS5, MATXS6, MATXS7, and MATXS8. There is no ENDF/B-V version of the 175x42 library. The materials included in each library vary with the application (MATXS12 is the largest, with 250 materials, because of the detailed fission-product cross sections provided). We have also processed 125 materials into ACE format for use with the MCNP Monte Carlo code.

These cross section libraries will soon be available from the Radiation Shielding Information Center (RSIC) at Oak Ridge.

DATA TESTING

An important part of the release of each new version of the ENDF/B libraries has been testing. This includes testing for proper use of the formats, testing the physical consistency of the data in the files, and testing against selected benchmarks. As part of the consistency testing, we have performed a detailed energy-balance study of ENDF/B-VI. Although there are still problems (including negative KERMA factors), it is overall a great improvement over ENDF/B-V, especially for the important structural materials Fe, Ni, Cr, and Cu. An example of the improvement obtained for Cr was shown in Figure 2. Table shows the materials that need the most improvement for the thermal range, fission range (.1 to 2 MeV), and fusion range (2 to 20 MeV). The entries in the thermal column have errors in photon energy production from 1 to 22%. The entries in the other two columns

Table 1: ENDF/B-VI Energy-Balance Problems

Thermal	Fission	Fusion
17-Cl-nat	42-Mo-nat	17-Cl-nat
42-Mo-nat	73-Ta-181	15-P-31
19-K-nat	63-Eu-151	11-Na-23
73-Ta-nat	79-Au-197	19-K-nat
31-Ga-nat	12-Mg-nat	42-Mo-nat
	19-K-nat	73-Ta-181
	67-Ho-165	74-W-182
	74-W-186	74-W-183
	74-W-184	74-W-184
	74-W-183	74-W-186
	31-Ga-nat	41-Nb-93
		22-Ti-nat
		63-Eu-151
		56-Ba-138

have negative KERMA values, values greater than twice the kinematic maximum, or values less than half of the kinematic minimum. Most of these materials were carried over from earlier ENDF/B versions.

We have also analyzed many CSEWG benchmarks, including the Los Alamos fast criticals (JEZEBEL, GODIVA, FLATOP, BIGTEN, etc.), the Argonne ZPR assemblies (6/7, 6/6a), some of the aqueous thermal criticals (ORNL1, ORNL2, ORNL10, L7, L8, L9), and some of the thermal lattices (TRX1, TRX2, BAPL1, BAPL2, BAPL3), using cross sections produced by TRANSX[9] with the ONEDANT and TWODANT transport codes. Table 2 shows the results for 2 bare criticals and the corresponding reflected criticals computed using 80 groups and P_3/S_{16} transport. Some numbers have improved and some have gotten worse, but the consistency between uranium and plutonium results is better. Table 3 shows results for the fast-breeder reactor mockups. These calculations used the simplified spherical homogeneous models for these assemblies together with standard corrections originally provided by ANL. Therefore, the actual numbers shouldn't be taken too seriously. However, the change from ENDF/B-V to ENDF/B-VI does suggest how the new libraries would change existing results for actual breeder-reactor calculations.

Table 2: Results for Godiva, Jezebel, Flattop-25, and Flattop-Pu

Assembly	Quantity	Experiment	C/E VI	C/E V
Godiva	k-eff	1.000±.002	.9983	.9990
	$\sigma_f(^{238}U)/\sigma_f(^{235}U)$	0.1643±.0018	.971	1.038
	$\sigma_f(^{233}U)/\sigma_f(^{235}U)$	1.59±.03	1.000	.986
	$\sigma_f(^{237}Np)/\sigma_f(^{235}U)$	.8516±.012	.960	1.044
	$\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$	1.4152±.014	.977	.985
Jezebel	k-eff	1.000±.002	.9989	.9982
	$\sigma_f(^{238}U)/\sigma_f(^{235}U)$	0.2133±.0023	.975	.961
	$\sigma_f(^{233}U)/\sigma_f(^{235}U)$	1.578±.027	1.000	.985
	$\sigma_f(^{237}Np)/\sigma_f(^{235}U)$	.9835±.014	.970	.979
	$\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$	1.4609±.013	.975	.966
Flattop-25	k-eff	1.000±.001	1.0030	1.0054
	$\sigma_f(^{238}U)/\sigma_f(^{235}U)$	0.1492±.0016	.977	1.034
	$\sigma_f(^{233}U)/\sigma_f(^{235}U)$	1.608±.003	.989	.975
	$\sigma_f(^{237}Np)/\sigma_f(^{235}U)$	.7804±.010	.979	1.054
	$\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$	1.3847±.012	.983	.989
Flattop-Pu	k-eff	1.000±.0014	1.0055	1.0050
	$\sigma_f(^{238}U)/\sigma_f(^{235}U)$	0.1799±.0020	.984	.974
	$\sigma_f(^{237}Np)/\sigma_f(^{235}U)$	.8561±.012	.987	1.002

Table 3: Results for ZPR-6/7 and ZPR-6/6A

Quantity	Experiment	Correction	C/E VI	C/E V
ZPR-6/7				
k-eff	1.000	−.0136	1.0070	.9997
$\sigma_f(^{238}U)/\sigma_f(^{239}Pu)$	0.02336±.0005	×1.030	1.023	1.021
$\sigma_f(^{235}U)/\sigma_f(^{239}Pu)$	1.061±.021	× .989	1.037	1.020
$\sigma_\gamma(^{238}U)/\sigma_f(^{239}Pu)$	0.1400±.003	×1.023	1.057	1.082
ZPR-6/6A				
k-eff	1.000	−.0043	1.0063	.9894
$\sigma_f(^{238}U)/\sigma_f(^{235}U)$	0.02411±.00072	×1.016	.968	.980
$\sigma_\gamma(^{238}U)/\sigma_f(^{235}U)$	0.1378±.0041	×1.011	1.003	1.042

Table 4: Results for the Spherical Homogeneous Uranium Assemblies

Assembly	Exp k	VI 187	V 187	VI 69	V 69
ORNL-1	1.00026	0.9969	1.0003	0.9984	1.0020
ORNL-2	0.99975	0.9967	1.0000	0.9982	1.0017
ORNL-10	1.00031	0.9972	1.0000	0.9979	1.0009
L7	1.0000	1.0050	1.0069	1.0015	1.0091
L8	1.0000	1.0044	1.0082		1.0134
L9	1.0000	1.0025	1.0060		1.0089

The performance of ENDF/B-VI for dilute homogeneous uranium assemblies is demonstrated by the results in Table 4. Group structures with 69 and 187 groups were used. The multiplication for the large assemblies (*e.g.*, ORNL-2) is now too small, and the bias towards excessive multiplication for small high-leakage assemblies (*e.g.*, L7) is only slightly reduced. A new version of ^{235}U that will greatly improve these results is in preparation by CSEWG. Results for two thermal lattice experiments are shown in Table 5. TRX-1 used metal rods, and BAPL-1 used oxide rods. Both were water moderated. Once again, simplified models were used, and the changes shown between ENDF/B-V and ENDF/B-VI are more important than the actual values quoted.

CONCLUSIONS

After a fairly long wait, ENDF/B-VI libraries for nuclear applications are now becoming available. The new evaluations have many improvements, but performance has worsened for a few applications. Users will have to make their own judgments about whether a change to the new libraries is justified for their own work.

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Table 5: Results for Thermal Lattice Experiments

Quantity	Experiment	C/E 69 VI	C/E 187 VI	C/E 69 V	C/E 187 V
TRX-1					
k-∞		1.1789	1.1822	1.1791	1.1829
k-eff	1.000	.9869	.9892	.9881	.9903
ρ^{28}	1.320±.021	1.033	1.023	1.040	1.032
δ^{25}	0.0987±.0010	.9993	1.004	1.013	1.018
δ^{28}	0.0946±.0041	1.021	1.036	1.021	1.037
C*	.797±.008	1.003	1.000	1.007	1.003
BAPL-1					
k-∞		1.1460	1.1462	1.1475	1.1483
k-eff	1.000	.9949	.9953	.9970	.9976
ρ^{28}	1.39±.01	1.003	1.003	1.007	1.010
δ^{25}	0.084±.002	.986	.993	1.000	1.006
δ^{28}	0.078±.004	.954	.970	.952	.968

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PROCESSING AND TESTING ENDF/B-VI WITH NJOY AND TRANSX

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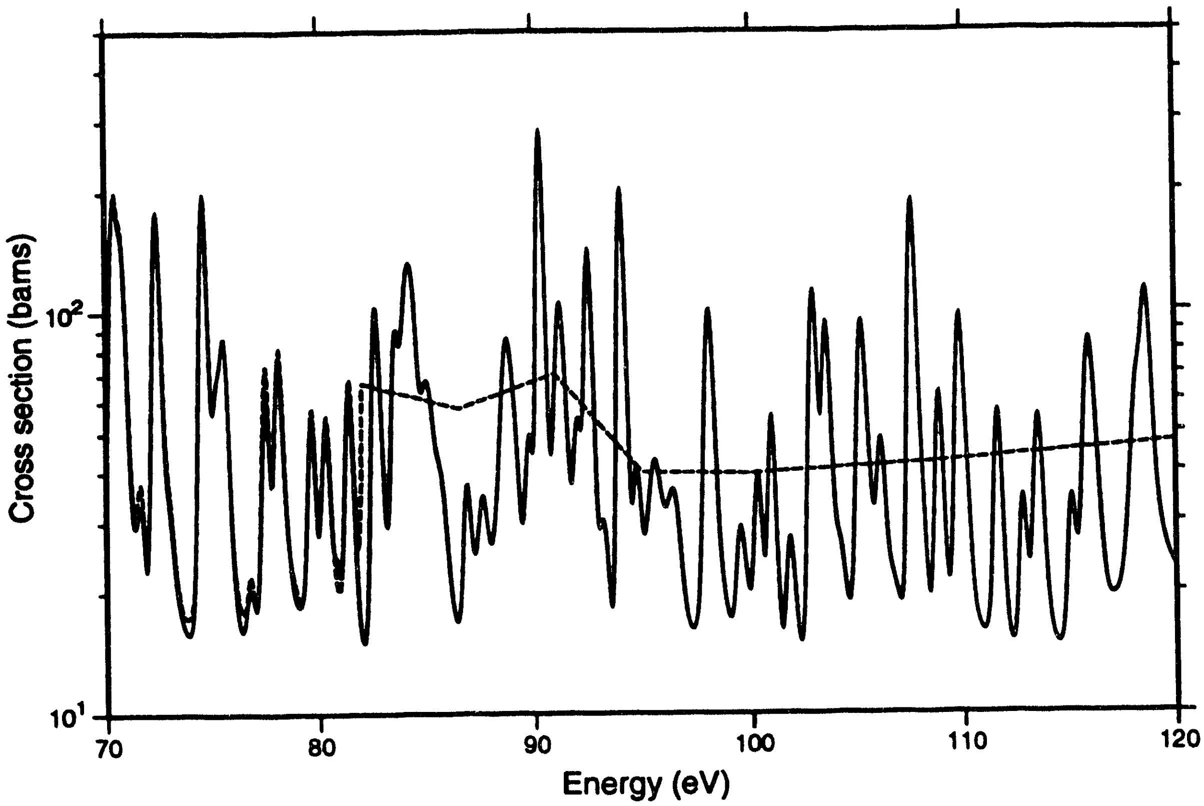
INTRODUCTION

- The NJOY Nuclear Data Processing System has been used to process ENDF/B-VI for multigroup and Monte Carlo libraries.
- New features of the ENDF-6 format
- Code modifications to handle the new evaluations
- Testing of code and data

ENDF/B-VI FEATURES

The ENDF/B-VI files contain a number of new features of interest to reactor physics:

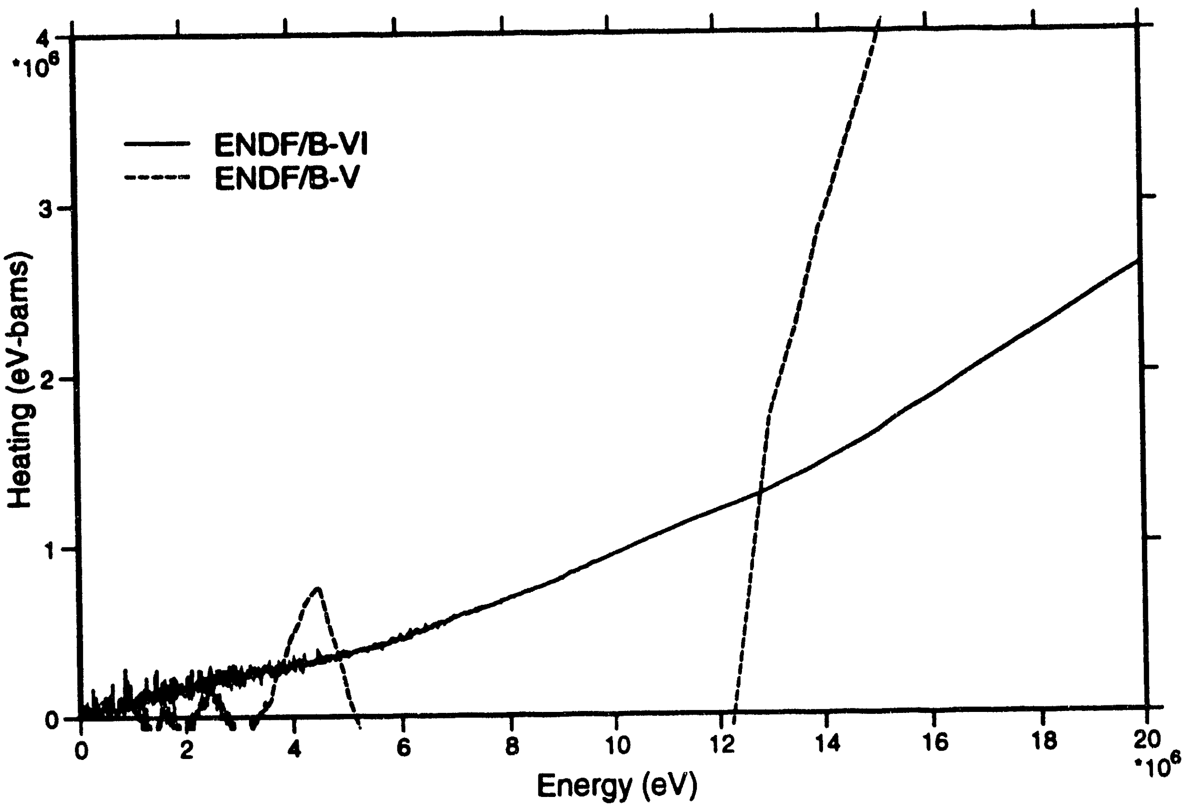
- ^{235}U, ^{238}U, and ^{239}Pu extended resonance ranges
  - ^{235}U: 82 eV extended to 2.25 keV
  - ^{238}U: 4 keV extended to 10 keV
  - ^{239}Pu: 300 eV extended to 1 keV
- Reich-Moore resonance parameters
  - Actinides
  - Structurals (Fe, Ni, Cr, Cu, Pb)
  - Better line shapes
  - No backgrounds
- Energy-angle distributions
  - Better high-energy transport
  - Better KERMA and damage



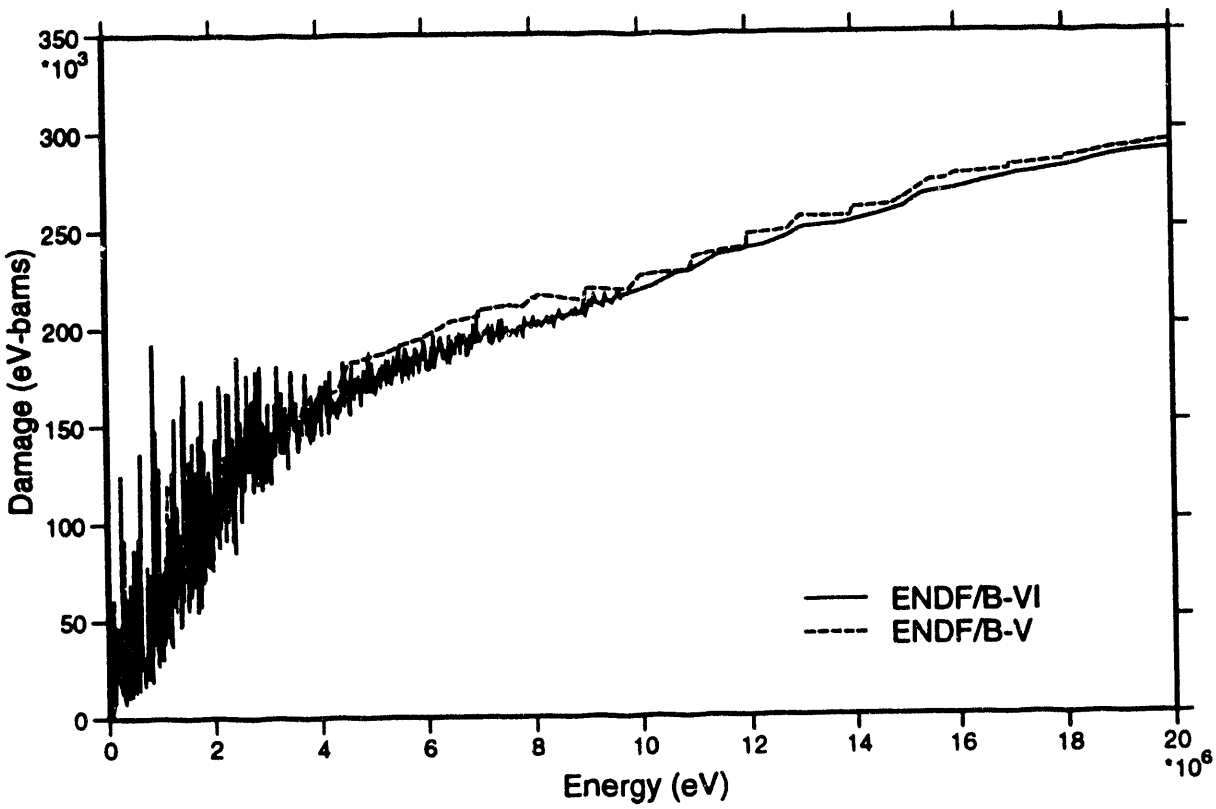
Total cross section for ^{235}U showing that part of the range that was given as unresolved in ENDF/B-V (dashed) is well resolved in ENDF/B-VI (solid).

FILE 6 BENEFITS FOR HEATING AND DAMAGE

- Isotopic evaluations are given.
- Explicit energy-angle distributions are given for emitted charged particles and recoil nuclei.
- Heating and damage can be computed directly without recourse to energy-balance arguments or built-in reaction models.
- ENDF/B-V required: *CP Energy = Available Energy − Neutron Energy − Photon Energy.*



KERMA comparison for chromium



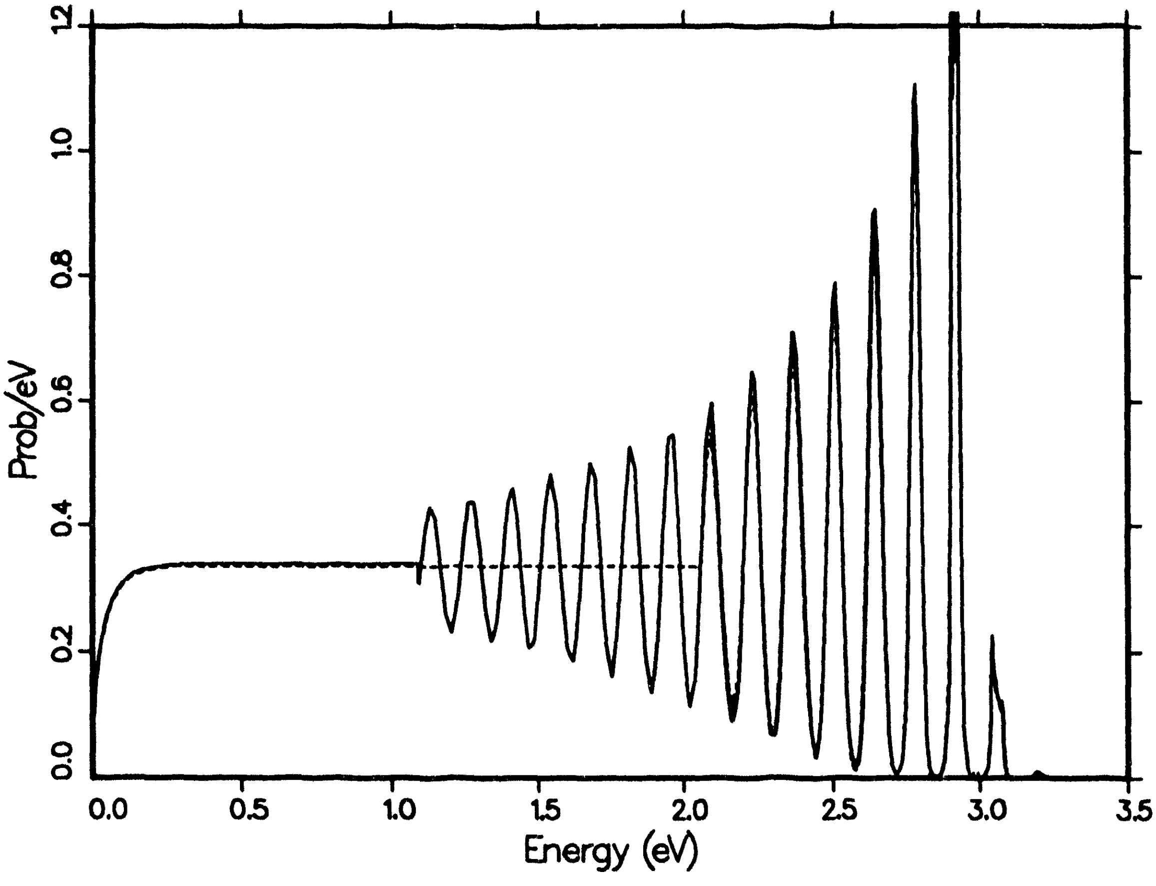
Damage comparison for chromium.

FILE 6 BENEFITS FOR TRANSPORT

- Coupled energy-angle distributions for secondary neutrons and outgoing particles in "File 6"
  - Kalbach systematics (LANL)
  - Legendre distributions in lab (ORNL)
  - Angle-energy distributions in lab (LLNL)
  - Phase-space distributions (LANL)
- Important for fusion and shielding applications
- Effects for fission-reactor problems remain to be evaluated.

THERMAL CHANGES

- File 7 format changes
- Conversion of old ENDF/B-III evaluations originally prepared by GAB.
- New LANL evaluations to improve accuracy and extend energy range using GA models.



Neutron spectrum from H in ZrH for an energy of 3.059 eV. The solid curve is for the new evaluation, and the dashed curve is for the older evaluation.

NJOY DEVELOPMENT

- NJOY has been gradually upgraded to handle the new features of the ENDF-6 format.
- This effort has been aided by users at other laboratories in the US and around the world.
- Much of the coding to handle the new resonance options was provided by Dunford (NNDC).
- Resonance reconstruction was tested by an international effort (IAEA, NEA Data Bank, ORNL).
- The File 6 energy-angle work as done was at Los Alamos
  - HEATR module to compute heating and damage
  - GROUPR module to compute multigroup transfer matrices for emitted neutrons, charged particles, and photons
  - ACER module for producing MCNP libraries (cooperation between T-2 and X-6).
- Latest steps consisted of getting NJOY and MCNP to work together properly for ENDF/B-VI data.
  - MCNP4A released.
  - Legendre distributions as used by ORNL are currently approximated by converting them to Kalbach form for sampling.

PROCESSED LIBRARIES

- NJOY 91.91 was used to process many of the materials of ENDF/B-VI.2 into several multigroup libraries in MATXS format.
  - MATXS10 is a 30x12 library that is compact and useful for high-energy problems where resonance effects are not important.
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  - MATXS12 is a 69x24 thermal reactor library with self-shielding and upscatter (250 materials, including fission products).
  - MATXS13 is a 187x24 multipurpose library for fusion, fast-reactor, and thermal-reactor problems.
  - MATXS14 is a 175x42 library for fusion neutronics studies.
  - ENDF/B counterparts: MATXS5, MATXS6, MATXS7, MATXS8.
- ACE-format library for MCNP (125 materials).

These cross section libraries will soon be available from the Radiation Shielding Information Center (RSIC) at Oak Ridge.

DATA TESTING

An important part of the release of each new version of the ENDF/B libraries has been testing:

- Testing for proper use of the formats
- Testing the physical consistency of the data in the files
- Testing against selected benchmarks

CONSISTENCY TESTING
ENERGY BALANCE

As part of the consistency testing, we have performed a detailed energy-balance study of ENDF/B-VI:

- ENDF/B-VI is overall a great improvement over ENDF/B-V, especially for the important structural materials Fe, Ni, Cr, and Cu.
- There are still problems (including negative KERMA factors)

Energy-Balance Problems for ENDF/B-VI:

Summary of the materials that need the most improvement for the thermal range, fission range (.1 to 2 MeV), and fusion range (2 to 20 MeV). The entries in the thermal column have errors in photon energy production from 1 to 22%. The entries in the other two columns have negative KERMA values, values greater than twice the kinematic maximum, or values less than half of the kinematic minimum. Most of these materials were carried over from earlier ENDF/B versions.

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BENCHMARK TESTING

- We have analyzed many CSEWG benchmarks:
  - Los Alamos fast criticals (JEZEBEL, GODIVA, FLATOP, BIGTEN, etc.)
  - ANL ZPR assemblies (6/7, 6/6a)
  - Some of the aqueous thermal criticals (ORNL1, ORNL2, ORNL10, L7, L8, L9)
  - Some of the thermal lattices (TRX1, TRX2, BAPL1, BAPL2, BAPL3)
- Using cross sections produced by TRANSX
- Using the ONEDANT and TWODANT transport codes

Results for 2 LANL bare criticals and the corresponding reflected criticals computed using 80 groups and P_3/S_{16} transport. Some numbers have improved and some have gotten worse, but the consistency between uranium and plutonium results is better.

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Jezebel	k-eff	1.000±.002	.9989	.9982
	$\sigma_f(^{238}U)/\sigma_f(^{235}U)$	0.2133±.0023	.975	.961
	$\sigma_f(^{233}U)/\sigma_f(^{235}U)$	1.578±.027	1.000	.985
	$\sigma_f(^{237}Np)/\sigma_f(^{235}U)$	.9835±.014	.970	.979
	$\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$	1.4609±.013	.975	.966
Flattop-25	k-eff	1.000±.001	1.0030	1.0054
	$\sigma_f(^{238}U)/\sigma_f(^{235}U)$	0.1492±.0016	.977	1.034
	$\sigma_f(^{233}U)/\sigma_f(^{235}U)$	1.608±.003	.989	.975
	$\sigma_f(^{237}Np)/\sigma_f(^{235}U)$	.7804±.010	.979	1.054
	$\sigma_f(^{239}Pu)/\sigma_f(^{235}U)$	1.3847±.012	.983	.989
Flattop-Pu	k-eff	1.000±.0014	1.0055	1.0050
	$\sigma_f(^{238}U)/\sigma_f(^{235}U)$	0.1799±.0020	.984	.974
	$\sigma_f(^{237}Np)/\sigma_f(^{235}U)$	.8561±.012	.987	1.002

Results for the fast-breeder reactor mockups. These calculations used the simplified spherical homogeneous models for these assemblies together with standard corrections originally provided by ANL. Therefore, the actual numbers shouldn't be taken too seriously. However, the change from ENDF/B-V to ENDF/B-VI does suggest how the new libraries would change existing results for actual breeder-reactor calculations.

Quantity	Experiment	Correction	C/E VI	C/E V
ZPR-6/7				
k-eff	1.000	−.0136	1.0070	.9997
$\sigma_f(^{238}U)/\sigma_f(^{239}Pu)$	0.02336±.0005	×1.030	1.023	1.021
$\sigma_f(^{235}U)/\sigma_f(^{239}Pu)$	1.061±.021	× .989	1.037	1.020
$\sigma_\gamma(^{238}U)/\sigma_f(^{239}Pu)$	0.1400±.003	×1.023	1.057	1.082
ZPR-6/6A				
k-eff	1.000	−.0043	1.0063	.9894
$\sigma_f(^{238}U)/\sigma_f(^{235}U)$	0.02411±.00072	×1.016	.968	.980
$\sigma_\gamma(^{238}U)/\sigma_f(^{235}U)$	0.1378±.0041	×1.011	1.003	1.042

Results for dilute homogeneous uranium assemblies using group structures with 69 and 187 groups.

Assembly	Exp k	VI 187	V 187	VI 69	V 69
ORNL-1	1.00026	0.9969	1.0003	0.9984	1.0020
ORNL-2	0.99975	0.9967	1.0000	0.9982	1.0017
ORNL-10	1.00031	0.9972	1.0000	0.9979	1.0009
L7	1.0000	1.0050	1.0069	1.0015	1.0091
L8	1.0000	1.0044	1.0082		1.0134
L9	1.0000	1.0025	1.0060		1.0089

Conclusions:

- The multiplication for the large assemblies (*e.g.*, ORNL-2) is now too small.
- The bias towards excessive multiplication for small high-leakage assemblies (*e.g.*, L7) is only slightly reduced.

A new version of ^{235}U that will greatly improve these results is in preparation by CSEWG.

Results for two thermal lattice experiments:

- TRX-1 used metal rods.
- BAPL-1 used oxide rods.
- Both were water moderated.
- Simplified models were used, and the changes shown between ENDF/B-V and ENDF/B-VI are more important than the actual values quoted.

Quantity	Experiment	69 VI	187 VI	69 V	187 V
TRX-1					
k-∞		1.1789	1.1822	1.1791	1.1829
k-eff	1.000	.9869	.9892	.9881	.9903
ρ^{28}	1.320±.021	1.033	1.023	1.040	1.032
δ^{25}	0.0987±.0010	.9993	1.004	1.013	1.018
δ^{28}	0.0946±.0041	1.021	1.036	1.021	1.037
C*	.797±.008	1.003	1.000	1.007	1.003
BAPL-1					
k-∞		1.1460	1.1462	1.1475	1.1483
k-eff	1.000	.9949	.9953	.9970	.9976
ρ^{28}	1.39±.01	1.003	1.003	1.007	1.010
δ^{25}	0.084±.002	.986	.993	1.000	1.006
δ^{28}	0.078±.004	.954	.970	.952	.968

Table give C/E values.

CONCLUSIONS

After a fairly long wait, ENDF/B-VI libraries for nuclear applications are now becoming available. The new evaluations have many improvements, but performance has worsened for a few applications. Users will have to make their own judgments about whether a change to the new libraries is justified for their own work.

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