An Introduction to the ENDF Formats

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Abstract

The ENDF Evaluated Nuclear Data Formats are used all over the world to encode nuclear data evaluations for use in research and nuclear technology. These lectures provide an introduction to the formats and how they are used in modern compilations of nuclear data. The introduction is also available on line, including more details and exercises, at *http://t2.lanl.gov/endf*.

1 What's In A Name?

The ENDF formats were originally developed for use in the US national nuclear data files called ENDF/B (the Evaluated Nuclear Data Files). These files went through various versions with names like ENDF/B-III, ENDF/B-IV, and ENDF/B-V. The formats were upgraded with each version to handle new features, for example, the extension from the original upper limit of 15 MeV to 20 MeV, the addition of photon production information, the introduction of new resonance formats, or the addition of charged-particle data. As ENDF/B-VI was being prepared, it was noted that the ENDF formats were coming into wide use around the world, including the JEF files in Europe, the JENDL files in Japan, and the BROND files in Russia. It was decided to decouple the ENDF formats from the ENDF/B libraries in order to make this international use easier. Therefore, we now refer to the "ENDF-6 format" to distinguish it from the "ENDF/B-VI library."

Control over the ENDF formats has been retained by the US Cross Section Evaluation Working Group (CSEWG), and the format specifications are published through the National Nuclear Data Center (NNDC) at the Brookhaven National Laboratory (*http://www.nndc.bnl.gov*).

2 What Are They For?

The uses of the ENDF formats have also evolved over the years. The first few versions were largely intended for thermal-reactor applications. ENDF/B-IV and ENDF/B-V shifted the emphasis toward fast-reactor and fusion applications. For ENDF/B-VI, additional extensions have been made for charged-particle and accelerator applications. In recent years, the ENDF system has also gained a role as a mode of publication and archiving of basic low-energy nuclear physics data. In general,

- ENDF-format libraries are computer-readable files of nuclear data that describe nuclear reaction cross sections, the distributions in energy and angle of reaction products, the various nuclei produced during nuclear reactions, the decay modes and product spectra resulting from the decay of radioactive nuclei, and the estimated errors in these quantities.

- ENDF-format libraries are intended to be used for a wide variety of applications that require calculations of the transport of neutrons, photons, and charged-particles through materials, the enumeration of the interactions of this radiation with the materials and their surroundings, and the time evolution of the radioactivity associated with the nuclear processes.

Examples of uses for ENDF-based libraries include fission and fusion reactor calculations, shielding and radiation protection calculations, criticality safety, nuclear weapons, nuclear physics research, medical radiotherapy, radioisotope therapy and

diagnostics, accelerator design and operations, geological and environmental work, radioactive waste disposal calculations, and space travel calculations.

3 Nuclear Data Evaluations

The answer to the question "What are they for?" has several implications and consequences. The work that is required to generate a file in ENDF format is called "nuclear data evaluation."

The requirement that the data be in a computer readable format puts strict limits on what an evaluator can do. Not every fact or physical effect can always be squeezed into the constraints of the format. But the evaluator has to do his or her best to get a good representation of reality. When these constraints are too limiting, we can try to extend the formats.

The fact that these evaluated data files are intended to be used for applications also has consequences. The data must be "complete" in some sense; for example, all energies must be represented, even when the data don't cover them all. Otherwise, particle transport calculations would be impossible. Modern evaluations are done by combining the experimental data with nuclear model code calculations to extend or interpolate the available data.

In addition, the fact that these evaluated data files are intended to be used for applications adds factors of quality control, revision control, peer review, and data testing that might not be needed for other modes of publication.

4 National Nuclear Data Organizations

The difficulty of modern nuclear data evaluation, together with the requirements for quality control, revision control, peer review, and data testing that result from the goal of using the data files for calculations that could have major impacts on public health and safety, has led to putting all modern nuclear data evaluation work under the control of a few national and international agencies, including:

- The Cross Section Evaluation Working Group (CSEWG), which handles the US ENDF/B libraries and the ENDF format. It is coordinated through the National Nuclear Data Center (NNDC) at the Brookhaven National Laboratory (*http://www.nndc.bnl.gov*).

- The JEF and EFF Working Groups, now merged as the JEFF Working Group, which handles the Joint European File (JEF), the European Fusion File (EFF), and their forthcoming merger, JEFF. It is coordinated through the NEA Data Bank (*http://www.nea.fr*), a part of the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD).

- The Japanese Nuclear Data Committee (JNDC), which handles the Japanese Evaluated Nuclear Data Library (JENDL). It is coordinated through the Nuclear Data Center at the Japan Atomic Energy Research Institute (JAERI) (*http://cracker.tokai.jaeri.go.jp*).

Each of these organizations has adopted the ENDF-6 format as the common method for publishing their nuclear data libraries, but each has been able to define its own procedures for using the formats, reviewing the evaluations, and testing the results.

5 Basic Organization

An ENDF-format nuclear data library has an hierarchical structure by tape, material, file, and section. Each of these levels has a characteristic numerical identifier:

- An ENDF "tape" is a file that contains one or more ENDF materials. We are fond of the word "tape" out of respect for the history of computing. As an example, Tape 511 was the ENDF/B-V Standards Tape.

- MAT labels an ENDF material. In older versions of ENDF/B, these numbers were assigned as the evaluations were completed, *e.g.,* MAT1301 is H-1 and MAT1395 is U-235 for ENDF/B-V. For ENDF/B-VI, CSEWG moved to MAT numbers computed from the target Z and A. The numbers step by threes to allow for isomers, and a last two digits of 25 point to the lightest of the common isotopes. Thus, 125 is H-1, 128 is H-2, 2625 is Fe-54, 6153 is Pm-148m, and 9228 is U-235. For ENDF/B-VI Tape and MAT numbers, see the online Index to ENDF/B-VI Neutron Data (*http://t2.lanl.gov/cgi-bin/nuclides/endind*).

- MF labels an ENDF file. "Files" are usually used to store different types of data, thus:

  MF=1 contains descriptive and miscellaneous data,
  MF=2 contains resonance parameter data,
  MF=3 contains reaction cross sections vs energy,
  MF=4 contains angular distributions,
  MF=5 contains energy distributions,
  MF=6 contains energy-angle distributions,
  MF=7 contains thermal scattering data,
  MF=8 contains radioactivity data
  MF=9-10 contain nuclide production data,
  MF=12-15 contain photon production data, and
  MF=30-36 contain covariance data.

- MT labels an ENDF section. Sections are usually used to hold different reactions. For example, MT=1 is the total cross section, MT=2 is elastic scattering, MT=16 is the (n,2n) reaction., MT=18 is fission, and MT=102 is

radiative capture. The ever increasing scope of use for the ENDF-format data has led to a continual increase in the number and types of reactions that can be represented (see the online version of this introduction or an ENDF format manual for a list of MT values).

An ENDF tape contains one or more materials in increasing order by MAT. Each material contains several files in increasing order by MF. Each file contains several sections in increasing order by MT.

6 Reading An ENDF Tape

The keys to finding your way around on an ENDF tape are the MAT, MF, and MT numbers. On the ASCII versions of ENDF-format files, they appear in columns 67 to 75 of every "card" (we also use the term "card" out of respect for the history of computing–besides, some of us remember them real well!). The Fortran notation is I4,I2,I3. The MT value is zero to indicate the end of a section (SEND record), the MF value is zero to indicate the end of a file (FEND record), and the MAT number is zero to indicate the end of a material (MEND record). There is a special "tape ID" record at the beginning with the MAT value equal to the tape number, and a special tape-end record (TEND) with MAT=-1 at the end of the tape. Here is an example of the skeletal structure for a typical ENDF tape (the 5-digit sequence numbers in columns 76 to 80 have been omitted for clarity):

```
tape id                                    7777 0  0
start of MF1, MT451 (description)          1111 1451
  ...
SEND record                                1111 1  0
FEND record                                1111 0  0
start of MF2, MT151 (resonances)           1111 2151
 ...
SEND record                                1111 2  0
FEND record                                1111 0  0
start of MF3, MT1 (total cross section)    1111 3  1
  ...
SEND record                                1111 3  0
start of MF3, MT2 (elastic cross section)  1111 3  2
  ...
FEND record                                1111 0  0
MEND record                                   0 0  0
TEND record                                  -1 0  0
```

To find samples of this kind of structure, look at the "raw" ENDF files from ENDF/B-VI (*http://t2.lanl.gov/cgi-bin/nuclides/endind*).

7 Simple Cross Sections

Simple cross sections on ENDF files are examples of the general one-dimensional tabulation, or TAB1, record. As an example, here is the section for the (n,2n) reaction in natural silicon from ENDF/B-VI:

```
 1.400000+4 2.784400+1          0          0          0          01400 3 16
-8.473800+6-8.473800+6          0          0          1         121400 3 16
        12          2                                              1400 3 16
 8.778100+6 0.000000+0 1.000000+7 6.166000-3 1.100000+7 1.564000-21400 3 16
 1.200000+7 2.589000-2 1.300000+7 3.650000-2 1.400000+7 4.663000-21400 3 16
 1.500000+7 5.400000-2 1.600000+7 5.620000-2 1.700000+7 5.734000-21400 3 16
 1.800000+7 5.830000-2 1.900000+7 5.870000-2 2.000000+7 5.892000-21400 3 16
 0.000000+0 0.000000+0          0          0          0          01400 3  0
```

The first line is the HEAD record; it contains the ZA value (100*Z+A) and the AWR value (ratio of target mass to neutron mass). The second card starts the TAB1 record and contains the reaction Q value (-8.4738 MeV) and some counts. The third line contains some interpolation information. Finally, the rest of the record contains the tabulation given as energy, cross section pairs with energies in eV and cross sections in barns. Therefore, we can immediately read off the 14 MeV cross section of .04663 barns. The last line in the section is just the normal SEND record.

Note that this is an endothermic reaction (negative Q value), and it has a threshold energy of 8.7781 MeV. We can also compute the threshold from the Q value using the formula -Q*(AWR+1)/AWR, which gives 8.778131 MeV. It is important to have the first energy point greater than or equal to the computed threshold in some applications, and NJOY checks this.

To fully understand this example, we need to have a more formal definition for the contents of a cross section record, more details on how Q values are used (why are there two values given?), and a description of how interpolation is used to define cross section values between the grid points given. The formal description of the File 3 format is available in the online version of this introductions, and the entire ENDF-102 Format Manual can be downloaded from *http://www.nndc.bnl.gov*.

8 ENDF Interpolation

Many types of ENDF data are given as a table of values on a defined grid with an interpolation law to define the values between the grid points. Simple one-dimensional "graph paper" interpolation schemes and a special Gamow interpolation law for charged-particle cross sections are provided.

The function $y(x)$ is represented by a series of tabulated values, pairs of x and $y(x)$, plus a method for interpolating between input values. The pairs are ordered by increasing values of x. There will be NP pairs given. The complete region over

which x is defined is broken up into NR interpolation ranges. An interpolation range is defined as a range of the independent variable x in which a specified interpolation scheme can be use; *i.e.,* the same scheme gives interpolated values of $y(x)$ for any value of x within this range. The definitions of the quantities in the interpolation table follow:

NP: Number of pairs x, y given
NR: Number of interpolation ranges given
INT(m): interpolation scheme used in the mth range
NBT: pair index separating the mth and (m+1)th ranges

The allowed interpolation schemes are

INT=1: y is constant in x (constant, histogram)
INT=2: y is linear in x (linear-linear)
INT=3: y is linear in $\ln(x)$ (linear-log)
INT=4: $\ln(y)$ is linear in x (log-linear)
INT=5: $\ln(y)$ is linear in $\ln(x)$ (log-log)
INT=6: y obeys a Gamow charged-particle penetrability law

Interpolation code INT=1 (constant) implies that the function is constant and equal to the value given at the lower limit of the interval. Note that where a function is discontinuous (for example, when resonance parameters are used to specify the cross section in one range), the value of x is repeated and two different y values are given to make a discontinuity. As an example, the most common interpolation table in the ENDF/B files simply specifies that linear-linear interpolation is used throughout the range of x.

```
NR=1
NP=10
10    2
```

A more interesting example might be as follows:

```
NR=3
NP=10
2    2    6    5    10    1
```

which says that linear-linear interpolation is used between the first point (*e.g.,* the threshold) and the second point. Log-log interpolation is used between the second and fifth points, and histogram interpolation is used above the fifth point. For histogram interpolation, the value of x for the last point is used to define the end of the range of $y(x)$ and the y value is ignored.

9 Interpolation Problems

The limitation to "graph paper" interpolation schemes causes some problems for reactions that are a sum of processes with different characteristic shapes. The classic example of this is the total cross section at low energies. At zero temperature, the elastic cross section tends to be constant for many materials, and it can be represented well using linear-linear interpolation. But the radiative capture cross section usually goes like $1/v$, and it is best described using log-log interpolation. Clearly, the sum of these two reactions will be OK at the grid points, but values intermediate between the grid points cannot be calculated with either linear-linear or log-log interpolation.

For this reason, summation cross sections, such as MT=1 (total cross section), MT=4 (total inelastic), and sometimes MT=18 (total fission), must not be considered fundamental. They must always be reconstructed from the sum of their parts.

In the NJOY Nuclear Data Processing System, linearization takes place in the RECONR module. A new energy grid is chosen iteratively that will represent each fundamental cross section, such as MT=2 and MT=102 as described above, to some desired accuracy (*e.g.,* 0.1%). The total cross section is then regenerated on the new grid by adding up the parts.

10 Resonance Parameters

Except for the lightest isotopes, elastic, capture, fission, and sometimes inelastic cross sections show sharp peaks and valleys due to resonances. For the lighter isotopes, it is practical to include all the pointwise detail of the cross sections directly in File 3. Good examples of this approach include carbon, O-16, and Al-27 in the ENDF/B libraries.

For heavier isotopes, this approach would lead to very large files with tens of thousands of energy points. ENDF-format nuclear data evaluations combat this problem by using resonance cross section formulas to calculate the elastic, capture, and fission cross sections over a defined "resonance range." The parameters to use in these formulas are given in File 2 using MT=151. The ENDF-6 format allows the following options:

Single-Level Breit-Wigner
Multi-Level Breit-Wigner
Reich-Moore
Adler-Adler
Hybrid R-Function
Generalized R-Matrix

At higher energies and in heavier isotopes, the resonances get to be so closely spaced that they can no longer be resolved into separate peaks experimentally. In

this "unresolved range," the ENDF format provide three ways to provide average resonance parameters to be used with statistical models from resonance theory to compute the cross sections:

Energy-Independent Parameters
Energy-Independent Parameters with Energy-Dependent Fission
Energy-Dependent Parameters

The cross sections computed from the resonance formulas have to be combined with the cross sections given in File 3 to determine the evaluated cross sections. In most cases, the resonance energy range in File 3 for the resonance reactions (total, elastic, fission, capture) will either contain the value zero, or it will contain small positive and negative values to be added to the computed resonance cross sections as corrections. In a few unresolved cases, the unresolved resonance range in File 3 will contain the actual "infinitely dilute" cross section, which is to be multiplied by resonance shielding factors determined from the unresolved-resonance parameters.

11 Single-Level Breit-Wigner Resonances

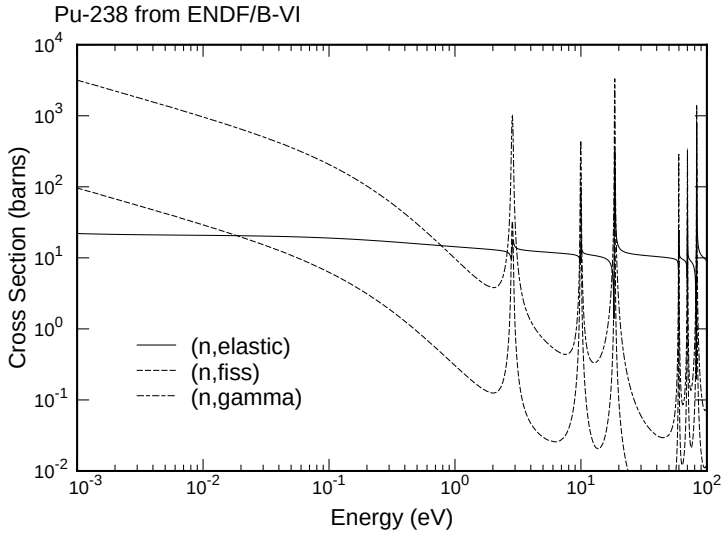
Earlier versions of the various evaluated data libraries made very heavy use of the Single-Level Breit-Wigner (SLBW) approach for the following reasons:

- it is easy to implement,
- it can use published resonance parameters,
- it can be Doppler broadened analytically, and
- it can be used analytically in reactor physics calculations.

However, it doesn't provide the best representation of the cross sections because of the neglect of multi-level and multi-channel effects–it can even produce unphysical negative cross sections for elastic scattering. Thus, some of the older files had corrections given as "File 3 backgrounds." Here is an shortened example of the SLBW resonance parameters for Pu-238 from ENDF/B-VI:

```
 9.423800+4 2.360045+2          0          0          1          09434 2151
 9.423800+4 1.000000+0          0          1          2          09434 2151
 1.000000-5 2.000000+2          1          1          0          09434 2151
 0.000000+0 9.309000-1          0          0          1          09434 2151
 2.360045+2 0.000000+0          0          0         96         169434 2151
-1.000000+1 5.000000-1 6.217000-2 1.581000-2 4.500000-2 1.360000-39434 2151
-4.000000-1 5.000000-1 4.670000-2 3.400000-4 4.500000-2 1.360000-39434 2151
 2.855000+0 5.000000-1 3.808600-2 7.470000-5 3.680000-2 1.211000-39434 2151
 9.975000+0 5.000000-1 3.721300-2 2.084000-4 3.024000-2 6.765000-39434 2151
 1.856000+1 5.000000-1 4.249000-2 3.490000-3 3.739000-2 1.610000-39434 2151
 5.980000+1 5.000000-1 3.930000-2 1.550000-3 3.480000-2 2.950000-39434 2151
 ...
```

The third line says that the resonance range extends from 1e-5 to 200 eV. The fourth line says that the target spin is 0 and the scattering length is .9309. This translates into a potential scattering cross section of $4\pi a^2 = 10.89$ barns. The parameters for 16 different resonances start on the sixth line: there is a negative-energy resonance at -10 eV with J=1/2, a total width of .06217 eV, a scattering width of .01581 eV, a capture width of .045 eV, and a fission width of .00136 eV. When these parameters are given to the RECONR module of the NJOY and Doppler broadened to 300K, the resulting cross sections are shown below.



12 Multi-Level Breit-Wigner Resonances

Some materials in both old and new libraries use the MLBW representation because the interference effects are too strong for the simpler SLBW representation. MLBW has the advantage of always being positive definite also. However,

- it doesn't handle materials with multi-channel effects,
- it is much more expensive to process than SLBW,
- it doesn't lend itself to analytic Doppler broadening, and
- it doesn't work well using analytic reactor physics methods.

The multi-level Breit-Wigner parameters use the same format in File 2 as the SLBW representation.

13 Reich-Moore Resonances

The Reich-Moore resonance representation (pronounced like "Rich-More") can handle both multi-level and multi-channel effects with good fidelity. No background corrections in File 3 are needed. Methods are now becoming available (the "multipole expansion") to allow analytic Doppler broadening and the direct use of Reich-Moore parameters in analytic reactor-physics methods. The Reich-Moore approach is used by the resonance analysis code SAMMY from ORNL, which is the current state of the art. Because of these factors, it has become the dominant choice for many of the new evaluations in ENDF/B-VI, including both fissionable isotopes with two fission channels like U-235, and nonfissionable materials like iron, copper, and lead.

```
 9.223501+4 2.330250+2          0          0          1          09228 2151
 9.223501+4 1.000000+0          0          1         12          09228 2151
 1.000000-5 4.000000+0          1          3          0          09228 2151
 3.500000+0 9.859600-1          0          0          1          39228 2151
 2.330200+2 9.859600-1          0          0        138         239228 2151
-1.000000+2 3.000000+0 1.145800-2 3.871290-2 1.229980-4 7.233640-29228 2151
-9.000000+1 4.000000+0 2.422100-6 3.680760-2 5.617020-2-2.168940-19228 2151
-4.297600+0 4.000000+0 7.164100-3 3.481860-2 3.192990-1-1.153500-19228 2151
-3.493400+0 3.000000+0 8.471500-8 3.780160-2-6.760010-3 1.298560-29228 2151
-1.504300+0 3.000000+0 8.519600-8 3.767610-2-7.010690-3 1.232090-29228 2151
-4.116100-1 3.000000+0 1.487500-4 2.984470-2-1.027260-3-1.554150-19228 2151
-1.942800-1 4.000000+0 5.044600-7 3.504170-2 1.989540-1-1.694210-39228 2151
 3.657500-5 4.000000+0 6.50520-11 2.984470-2-5.263430-4 9.645560-49228 2151
 2.819000-1 3.000000+0 4.439200-6 3.837130-2 1.065400-1-4.849860-39228 2151
 1.138900+0 4.000000+0 1.384200-5 4.069500-2-4.640000-6 1.093200-19228 2151
 2.036100+0 3.000000+0 9.358700-6 3.933000-2-7.736600-3-1.573500-39228 2151
 2.776700+0 4.000000+0 1.277200-6 3.887600-2 6.049200-2-4.250300-29228 2151
 3.156600+0 3.000000+0 2.432600-5 3.989600-2-7.995100-2 1.716100-29228 2151
 3.620800+0 4.000000+0 4.184000-5 3.764400-2-2.679600-2 2.849100-29228 2151
 4.850800+0 4.000000+0 7.560500-5 3.801700-2 4.666800-5-3.735100-39228 2151
 5.449700+0 4.000000+0 3.793200-5 3.920100-2-7.874000-2-3.612500-19228 2151
 6.209400+0 3.000000+0 1.654700-4 4.005100-2-1.079400-1 7.385900-29228 2151
 ...
```

In this case, the energy range being described is 1e-5 to 4.0 eV (other ranges are also given for higher energies up to 2.25 keV). The target spin is 3.5, and the scattering length of .98596 translates to a potential scattering cross section of 12.216 barns. The columns give resonance energy, J value, Γ_N, Γ_γ, Γ_{FA}, and Γ_{FB}. The existence of two fission widths is part of the explanation for the great fidelity of the Reich-Moore fit to the experimental data for U-235. There is only one ℓ value, ℓ=0, for this case.

14 Particle Distributions in Energy and Angle

In order to follow the transport of nuclear radiation through a material, it is important to know which secondary products are produced, the yield of each product, and how each product is distributed in energy and angle. The capabilities of the ENDF-format to represent this information about the products has evolved from fairly simple representations using Files 4 and 5 to the current rather complete capabilities found in File 6. In general, the cross section (in barns/steradian) for producing a particle can be written

$$\sigma(\mu, E, E') = \sigma(E) y_i(E) f_i(\mu, E, E')/2\pi \tag{1}$$

$$\int dE' \int d\mu f_i(\mu, E, E') = 1 \tag{2}$$

in terms of a cross section $\sigma(E)$, a yield $y(E)$, and a normalized distribution in initial energy E, final energy E', and cosine μ. The cross section is always given in File 3. The yield may be implicit as determined by the MT number, or in File 6, it may be given explicitly as integers for simple reactions or in noninteger form for the complex summation reaction MT=5. The distributions may be represented using three different approaches:

- For simple two-body reactions, E' can always be computed from kinematics, and it is only necessary to give $f(E, \mu)$. The function is given using File 4 or a special "law" in File 6. This option is used for elastic scattering (MT=2), neutron discrete-levels (MT=51-90), or neutron and particle discrete reactions (MT=50-90, 600-648, 650-698, *etc.*).

- For older evaluations (incident neutrons), it is often assumed that the secondary-neutron distribution can be represented as a product of an angular distribution and an energy distribution, *i.e.*, $f(E, \mu) \times g(E, E')$. The angular distribution is given in File 4 and the secondary-energy distribution is given in File 5. This approach is often used for MT=16, 18, or 91.

- For newer evaluations, the distributions are often given as fully coupled energy-angle distributions in File 6 using one of the several "laws" offered there. The use of File 6 is required for incident charged particles.

The File 5 representation is always used for fission in ENDF files. The neutrons are assumed to be emitted isotropically in the laboratory reference frame.

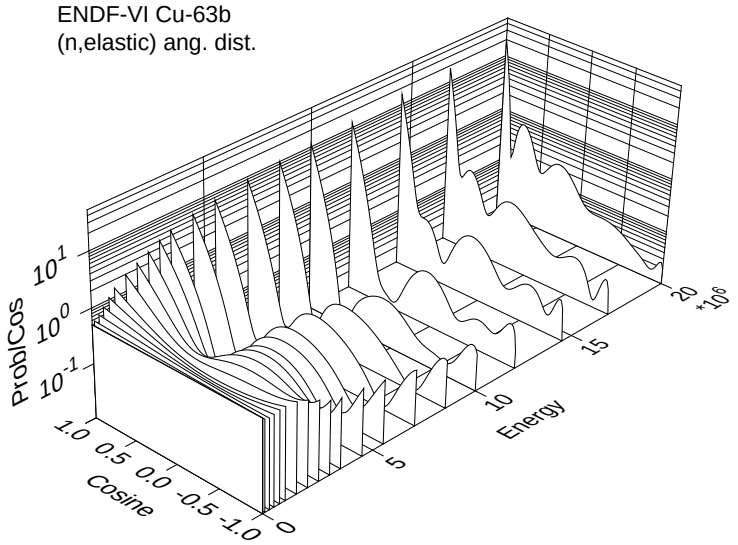
15 Angular Distributions

The angular distributions for two-body scattering reactions are given in either File 4 or a special "law" of File 6. The most common way of representing them uses

Legendre coefficients:

$$f(\mu, E) = \sum_{\ell=0}^{\mathrm{NL}} \frac{2\ell+1}{2} a_\ell(E) P_\ell(\mu) \tag{3}$$

At low energies for elastic scattering, the angular distribution will be isotropic in the center of mass reference frame. As the energy increases, the scattering typically becomes more and more forward peaked, and higher and higher Legendre orders will be needed. If the energy goes through a resonance, dramatic swings from forward scattering to backward scattering may be seen. The following figure below is typical.



At the highest energies, the scattering is basically the diffraction of a wave around a hard sphere, and the results are similar to diffraction patterns for light. The forward peak is like the bright central spot in a diffraction pattern, and the oscillations at larger angles are analogous to the fringes seen for light.

The following is an example of a section of File 4 for elastic scattering (^{63}Cu from ENDF/B-VI). The "1" in the fourth position of the first card indicates that the data are in the center-of-mass system. Cards 5 and 6 indicate that the scattering is isotropic at 1e-5 eV (the P_1 coefficient is zero). Cards 9 and 10 show anisotropy beginning to show up at 10 keV. The anisotropy gradually increases with energy until a Legendre order of 14 is needed to represent the angular distribution at 20 MeV.

```
2.906300+4 6.238900+1          0          1          0          02925 4  2
0.000000+0 6.238900+1          0          2          0          02925 4  2
0.000000+0 0.000000+0          0          0          1         222925 4  2
        22          2                                            2925 4  2
0.000000+0 1.000000-5          0          0          1          02925 4  2
0.000000+0                                                       2925 4  2
0.000000+0 2.530000-2          0          0          1          02925 4  2
0.000000+0                                                       2925 4  2
0.000000+0 1.000000+4          0          0          2          02925 4  2
3.214700-3 1.190800-4                                            2925 4  2
0.000000+0 1.000000+5          0          0          4          02925 4  2
3.619500-2 3.845600-3 3.661300-5 0.000000+0                      2925 4  2
0.000000+0 3.000000+5          0          0          4          02925 4  2
7.500000-2 1.800000-2 4.000000-4 0.000000+0                      2925 4  2
0.000000+0 5.000000+5          0          0          4          02925 4  2
1.200000-1 5.500000-2 2.550000-3 1.200000-4                      2925 4  2
0.000000+0 7.500000+5          0          0          4          02925 4  2
1.730000-1 1.070000-1 1.300000-2 2.730000-3                      2925 4  2
0.000000+0 1.000000+6          0          0          6          02925 4  2
2.258400-1 1.602700-1 3.980500-2 1.286300-2 1.560800-5 0.000000+02925 4  2
0.000000+0 1.500000+6          0          0          6          02925 4  2
2.738500-1 2.188700-1 9.602200-2 3.370000-2 1.499300-4 0.000000+02925 4  2
 ...
0.000000+0 2.000000+7          0          0         14          02925 4  2
8.105400-1 6.500300-1 5.507300-1 4.828500-1 4.177800-1 3.523300-12925 4  2
2.899200-1 2.378900-1 1.840200-1 1.207600-1 6.096100-2 2.102800-22925 4  2
4.210300-3 0.000000+0                                            2925 4  2
                                                                 2925 4  0
```

For formal specifications for File 4, see the online version or an ENDF Format Manual.

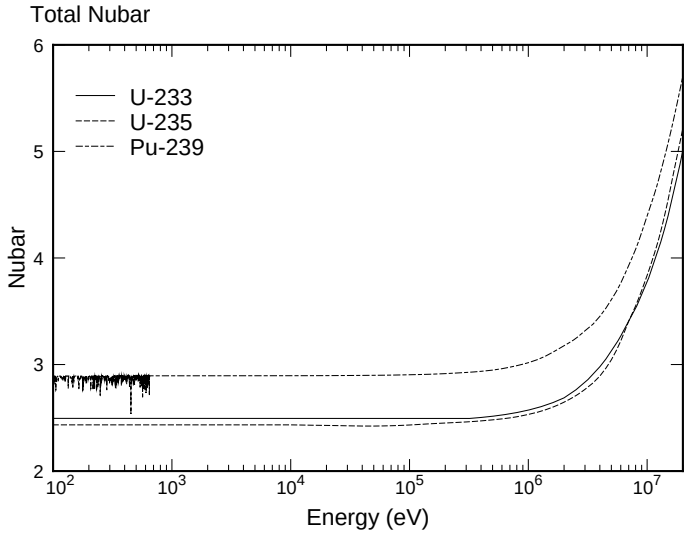
16 Fission Neutron Yields

The number of neutrons produced in a fission reaction is highly important for fission reactors, explosive fission devices, and the safe storage and processing of fissionable materials. It is an energy-dependent quantity, and it has both prompt and delayed components. Fission neutron yields, or "nu-bar's," are given in three special sections of File 1:

MT=452, total neutrons per fission,
MT=456, prompt neutrons per fission, and
MT=455, delayed neutrons per fission and time constants.

Upon fission, the excited nucleus formed by the target and the projectile breaks up into two roughly equal "fission fragments" and a number of additional neutrons (1, 2, 3, 4, ...). It is the average over all these possible break-up modes that gives the "prompt" nu-bar (*e.g.,* 2.5). Some of the excited fission fragments emit additional neutrons as they decay over the next second or so; these are called "delayed neutrons."

Prompt nu-bar tends to be constant at low energies, and then it gradually increases as the energy increases. In earlier times, this increasing trend was often represented as a simple polynomial (approximately linear) in energy E. More detailed analysis shows more structure associated with second- and third-chance fission processes. Some evaluations even have structure at very low energies. Therefore, most modern evaluations use a detailed tabulated function to represent the energy dependence of the fission nu-bar. The figure below shows a few examples.



The following example is U-233 from ENDF/V-VI. The "2" in the fourth field on the first line says that the nu-bar data are given in tabulated form. The actual tabulation of total nu-bar *vs* energy starts on line 4.

```
9.223300+4 2.310430+2          0          2          0          09222 1452
0.000000+0 0.000000+0          0          0          1         109222 1452
        10          2                                           9222 1452
1.000000-5 2.494700+0 2.530000-2 2.494700+0 3.200000+5 2.494000+09222 1452
2.000000+6 2.687400+0 4.500000+6 3.052000+0 6.000000+6 3.268000+09222 1452
6.500000+6 3.340900+0 1.400000+7 4.270400+0 1.500000+7 4.393800+09222 1452
2.000000+7 5.013500+0                                       9222 1452
                                                            9222 1  0
```

The formats for prompt nu-bar and total nu-bar are the same. The format for delayed nu-bar has an additional LIST record giving the decay constants that are used to describe the time history of delayed neutron emission. See the online version or an ENDF manual for the formal specifications.

17 Simple Emission Spectra

In many ENDF-format evaluations, the spectra of neutrons emitted in a reaction are given as independent of angle using File 5. This is reasonable in some cases, but at higher neutron energies there is often a strong correlation between the energy distribution and the angle of emission. Correlated energy distributions can be described using File 6. File 5 allows for several different representations of the spectra of outgoing neutrons:

LF=1, Arbitrary tabulation function A set of incident energies is given, and a normalized tabulated distribution is given at each incident energy. This option is very general, and it is the modern choice for detailed fission spectra, such as those produced with the Los Alamos Standard Model.

LF=5, General evaporation spectrum Here a temperature-like quantity θ is tabulated against incident energy E, and a single spectrum is given as a function of E'/θ. The only current application for this law is to give six different delayed-neutron spectra for six time groups, and θ is just set to one. See MF=5, MT=455 in fissionable materials.

LF=7, Simple Maxwellian fission spectrum This is a fairly old model for the distribution of neutrons emitted by fission, but it is still in fairly wide use. The effective temperature θ is tabulated against the incident energy E, and the range of final energies allowed is from zero to $E - U$.

LF=9, Evaporation spectrum This is the kind of shape that would be expected if the the compound nucleus during a reaction had lots of time to come into equilibrium before the neutron was emitted, and it may not be a bad approximation for fairly low incident energies (a few MeV). At higher energies, preequilibrium emission begins to change the shape away from the evaporation form, but this is also the region where energy-angle correlation begins to be important. Once again, the effective temperature of the compound system θ is tabulated against the incident energy E, and the range of final energies allowed is from zero to $E - U$.

LF=11, Energy-dependent Watt spectrum The Watt spectrum is a more sophisticated representation of fission than the simple Maxwellian law described above. Two functions $a(E)$ and $b(E)$ are tabulated in File 5, and the range of final energies allowed is from zero to $E - U$.

LF=12, Madland-Nix fission spectrum This is another sophisticated fission model based on two constants EFL and EFM, the average kinetic energies of the light and heavy fragments, respectively, and a temperature-like quantity TM that is given as a function of incident energy E.

18 Energy-Angle Distributions

The old-fashioned approach of representing an energy-angle distribution as a product of an angular distribution (MF=4) and a secondary energy distribution (MF=5) is no longer adequate for modern evaluations. For incident energies larger than a few MeV, preequilibrium effects begin to appear, which lead to a coupling between the emission angle and the emission energy. In addition, it is sometimes necessary to have detailed information about the emission of other particles than neutrons, for example, the protons from an (n,n′p) reaction. The combination of File 4 and File 5 only allows one particle to be represented. Most modern evaluations are made with the help of a nuclear model code, such as GNASH, to complete the ranges of energy and angle that are not available from experiment. These codes follow all reaction channels and emitted particles, and it is important to have a way to include all the code results into the ENDF format. It was these kinds of arguments that led to the development of File 6 as introduced for the ENDF-6 format.

A section of File 6 is divided into subsections, one for each product. The products can include neutrons and light charged particles ordered by ZA, followed by residual nuclei ordered by ZA, followed by photons, followed by electrons:

```
[MAT, 6, MT/ ZA, AWR, 0, LCT, NK, 0] HEAD
[MAT, 6, MT/ ZAP, AWP, LIP, LAW, NR, NP/ Eint/ Y(E)] TAB1
    law-dependent data for product 1
    ---------------------------------
    repeat TAB1 and law-dependent data
    for the rest of the NK products
    ---------------------------------
[MAT, 6, 0/ 0.0, 0.0, 0, 0, 0, 0] SEND
```

The individual products are identified by ZAP, AWP, and LIP (to allow for isomers). The different representations allowed in a subsection are defined by the value of LAW:

LAW=0, Unknown distribution;
LAW=1, Energy-angle distribution;
LAW=2, Two-body angular distribution;
LAW=3, Isotropic two-body distribution;
LAW=4, Recoil distribution for a two-body reaction;
LAW=5, Charged-particle elastic scattering;
LAW=6, N-body phase-space distribution; and
LAW=7, Laboratory angle-energy law.

19 Kalbach Distributions

File 6 with LAW=1 provides for an especially elegant and compact representation of coupled energy-angle distributions for neutrons and charged particles developed by

Kalbach based on systematics. The general shape of the function was later shown to come directly out of preequilibrium theory by Chadwick. The Kalbach distribution for a reaction of the form

$$A + a \rightarrow C \rightarrow B + b \tag{4}$$

can represented by

$$f(\mu_b, E_a, E_b) = f_0(E_a, E_b) \left\{ \frac{a}{\sinh(a)} [\cosh(a\nu_b) + r \sinh(a\mu_b)] \right\} \tag{5}$$

where $r(E_a, E_b)$ is the precompound fraction as given by the evaluator, and $a(E_a, E_b)$ is a simple parameterized function that depends on emission energy, incident energy, and particle type.

Note that if r and a are small, the distributions are basically isotropic. This corresponds to emission from a well equilibrated compound system. As r increases, the distribution becomes more forward peaked. Because r tends to increase with the emission energy, we see forward peaking for high particle energies and isotropy for low particle energies. This is the effect that cannot be represented using the combination of File 4 and File 5.

The following is an example taken from the ENDF/B-VI evaluation for ^{27}Al. It shows the (n,2n) neutron emission distribution for an incident energy of 40 MeV. Each line contains two triplets: E', $f(E, E')$, and $r(E, E')$. Note the increase in r with increasing E'.

```
0.000000+0 4.000000+7          0          1        198         661325 6 16
0.000000+0 1.463653-8 2.774204-4 3.609812+4 2.663865-8 2.908748-41325 6 16
7.219623+4 3.562323-8 3.043292-4 1.203271+5 4.369308-8 3.222684-41325 6 16
1.684579+5 5.047795-8 3.402077-4 2.165887+5 5.644923-8 3.581469-41325 6 16
...
4.091120+6 5.029411-8 8.229791-3 4.572428+6 4.580329-8 1.160481-21325 6 16
5.053736+6 4.182685-8 1.619895-2 5.535044+6 3.824886-8 2.245802-21325 6 16
6.016353+6 3.515369-8 3.071985-2 6.497661+6 3.237198-8 4.158862-21325 6 16
6.978969+6 2.981761-8 5.597233-2 7.460277+6 2.765722-8 7.438953-21325 6 16
7.941585+6 2.592982-8 9.753015-2 8.422894+6 2.461467-8 1.260132-11325 6 16
8.904202+6 2.364081-8 1.603900-1 9.385510+6 2.298213-8 2.006077-11325 6 16
9.866818+6 2.272568-8 2.456513-1 1.034813+7 2.276480-8 2.963706-11325 6 16
1.082943+7 2.312538-8 3.510374-1 1.131074+7 2.376400-8 4.081861-11325 6 16
1.179205+7 2.465185-8 4.658113-1 1.227336+7 2.581909-8 5.226821-11325 6 16
1.275467+7 2.726533-8 5.776199-1 1.323598+7 2.899395-8 6.296106-11325 6 16
1.371728+7 3.098722-8 6.779314-1 1.419859+7 3.324436-8 7.218539-11325 6 16
1.467990+7 3.557207-8 7.621944-1 1.516121+7 3.833738-8 7.979111-11325 6 16
...
2.430606+7 1.964407-9 9.958412-1 2.478737+7 0.000000+0 0.000000+01325 6 16
```

The Kalbach representation is used in many of the Los Alamos evaluations in ENDF/B-VI, and it is an important factor in the new high-energy libraries to 150 MeV now being constructed for accelerator applications at several laboratories in the world.

20 What Else Do You Need to Know?

This written introduction to the ENDF formats is necessarily brief. More information will be found in the online version, or by getting the full ENDF Format Manual and using it to study a variety of existing evaluations. Some important omissions include unresolved resonances, formats for photon production, fission product yields, activation data, the details of reaction naming, charged-particle formats, and data covariances.