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Abstract

ENDF is a format for storing nuclear data. It is widely used in nuclear industries and academia, but it not-at-all human-readable, despite being written in plain text. This text is written as a quick start guide for the average physicist; it assumes the reader has some understanding of nuclear reactions, but has never seen an ENDF file.

A gentle introduction into ENDF

1 MT and MF numbers

1. For each nuclide in the Segré chart, in its excited or ground state;
2. Multiple reactions can occur upon impact with a variety of projectiles;
3. And each reaction must be characterized by its probability distribution with respect to
    - angle,
    - energy,
    - underlying mechanism,
    - ... etc.
    - and covariance matrix (uncertainty) of each

To deal with point 1, an identifying number $1 \leq$`MAT`≤ 9999 is assigned to each isomer.

To deal with point 2, an identifying number $1 \leq$`MT`≤ 999 is assigned to each reaction. A different MT number is assigned for each unique combination of incoming projectile + out-going particle. The full MT number table can be found in Appendix B of the ENDF manual.

Dealing with point 3 is more subtle. Instead of storing its probability of reaction(or reaction cross-section) as a scalar function with n input dimensions, where $n > 3$ (mechanism, angle, energy, decay time, etc.), which is going to be sparsely populated anyways, they are recorded as 1-dimensional lists of numbers (or, in the case of covariance matrices, 2-dimensional matrices of numbers), each list is given a number $1 \leq$`MF`≤ 40. For example, MF=3 records the reaction cross-section $\sigma(E)$ as a function of energy E (integrated over all solid angle); MF=4 records the probability of emission $P(\theta)$ as a function of angle θ (fixed at $E = \frac{1}{40}eV$), etc. See Table 2 below for more details.

MF\MT	1(n,total)	2(z,z$_0$)	4(z,n)	151(n,RES)
1(General information)				Resonance region (*)
2(Resonance param. data)				effective scattering radius
3(Reaction cross-section)	$\sigma_T(E)$	$\sigma_{\text{elas}}(E)$	$\sigma_{(n,2n)}(E)$	$\sigma(E_{RES})$ (a scalar)
4(Angular distributions)	N/A	$P(\theta_{\text{scattered n.}})$	$P(\theta_{\text{expelled n.'s}})$	N/A
5(Energy distributions)	N/A	$P(E_{\text{scattered n.}})$	$P(E_{\text{expelled n.'s}})$	N/A

Table 1: For each isomer, each MT number denotes a possible reaction, and each MF number denotes what information is required to characterize this reaction.

See footnote[1] for explanation on * in Table 1.

A more tangible example is listed as follows: to get the angular distribution of non-elastically scattered neutrons from ^{1}H, we should look for

1. MAT=1001,
2. MT=3 (non-elastic neutron cross-section),
3. MF=4 (Angular distributions for emitted particles).

The complete MF table is included below for the reader's reference.

MF	Description
1	General information
2	Resonance parameter data
3	Reaction cross-sections
4	Angular distributions for emitted particles
5	Energy distributions for emitted particles
6	Energy-angle distributions for emitted particles
7	Thermal neutron scattering law data
8	Radioactivity and fission-product yield data
9	Multiplicities for radioactive nuclide production
10	Cross-sections for radioactive nuclide production
12	Multiplicities for photon production
13	Cross-sections for photon production
14	Angular distributions for photon production
15	Energy distributions for photon production
23	Photo- or electro-atomic interaction cross-sections
26	Electro-atomic angle and energy distribution
27	Atomic form factors or scattering functions for photo-atomic interactions
28	Atomic relaxation data
30	Data covariances obtained from parameter covariances and sensitivities
31	Data covariances for nu(bar)
32	Data covariances for resonance parameters
33	Data covariances for reaction cross-sections
34	Data covariances for angular distributions
35	Data covariances for energy distributions
39	Data covariances for radionuclide production yields
40	Data covariances for radionuclide production cross-sections

Table 2: The complete MF number table as copied from the ENDF manual's "Table 3"[1], for the reader's reference. Some values are unused as they are deprecated.

Therefore each ENDF "File" must have an MAT number, an MT number, and an MF number associated with it.

2 A simple illustration of the typical ENDF database layout

Perhaps the most misleading word in the whole ENDF nomenclature is the word "file": In the modern days of computing, it should really be called a "section within a text file"

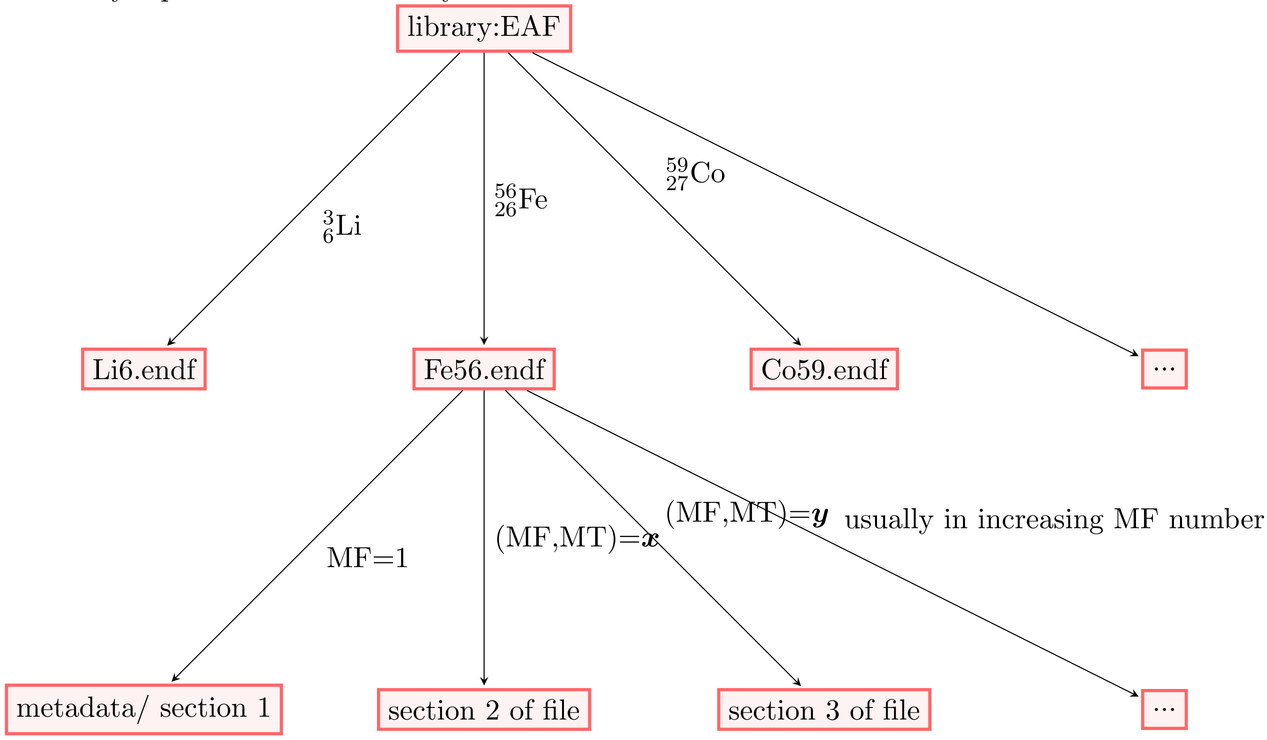
[1] 'Resonance parameters that can be used to calculate cross sections at different temperatures in the resolved and unresolved energy regions.'[1]Appendix B, which is directly linked here.

to prevent confusion.

As stated in section 0.4.2 of [1]: A "File" refers to 'a block of data in an evaluation that describes a certain data type'.

Here, each 'data type' refers to an aspect of a nuclear reaction:
e.g. for the case of inelastic neutron scattering, the angular distribution of the outgoing neutron is a 'data type'.
(MT=51 (n,n') , MF=6 (energy-angle distributions for emitted particles))

Concatenating multiple ENDF "Files" together gives a computer-readable text file. From this point onwards, you can safely assume that, whenever the word file appears without quotation marks, I am using the universal definition of a regular text file, whose end-of-file byte position is indicated by a EOF character.



The layout of the EAF library is illustrated above. The root (library:EAF) is the folder containing everything.

Inside the folder are files, named after each isotope. Each file contains some data about that element; but it rarely contains data of all 999 MT numbers × 40 MF numbers.

Inside each file are sections, each corresponding to a set of MT and MF numbers.

This is not the only way to organize the files and folders.

- ENDF files don't always have to end in ".endf"; as long as it is written in the same format as specified by [1] in ASCII text, it can be read by `OpenMC/pyne`.
- each file shown above actually consist of a collection of the ENDF "Files" referenced by section 0.4.2 of [1], concatenated together.
- One can even concatenate all of the files in the 2nd layer into a single, very long (hundreds of MB's) text file with an arbitrary file extension.
- Some of the more comprehensive libraries (e.g. JEFF) records data across a large range of MT numbers; therefore they may split their libraries into multiple roots. For example, the 2017 release of TENDL splits it into seven roots, sorted by projectile types:
  - `library:TENDL/d`
  - `library:TENDL/n`
  - `library:TENDL/g` (gamma)
  - `library:TENDL/he3`

- library:TENDL/he4
- library:TENDL/p
- library:TENDL/t

instead of a single root as shown in the EAF example above.

- You may also see the words "ACE" file frequently occurring in the context of ENDF data. This stands for A Compact ENDF, which is the format that is fed into MCNP simulations. (Normal ENDF files are too big to be used in MCNP.)

3 Using ENDF files

The purpose of this text is not to provide the technical know-how of designing and writing ENDF files; but to simply give the average nuclear data user, who has no intention of becoming a nuclear cross-section evaluator, an idea of how to use the publicly available ENDF files. Therefore this section will introduce the easiest approach(es) to extracting data from ENDF files.

3.1 The python approach

Simply download an archive of ENDF files from any official website of your choice, unzip it into your working directory, and read it into memory using a nuclear data management package of your choice (`openmc` or `pyne`). These will then be saved as instances of a class(e.g. an `openmc.data.endf.Evaluation` object), where values such as $\sigma(E)$ will be saved as its attributes.

References

[1] A Trkov, M Herman, DA Brown, et al. ENDF-6 formats manual. Data Formats and Procedures for the Evaluated Nuclear Data Files ENDF/B-VI and ENDF/B-VII, National Nuclear Data Center Brookhaven National Laboratory, Upton, NY, pages 11973–5000, 2012.