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Nuclear Evaluation Methods

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Intended for

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Contents :

Chapter 1 : Introduction to nuclear data evaluation

Chapter 2 : Evaluated nuclear data libraries

Chapter 3 : ENDF Format

Practical Work 1 : ENDF Libraries

Practical Work 2 : PREPRO CODES

Practical Work 3 : JANIS CODE

Chapter 1

Introduction to nuclear data evaluation

1. Nuclear data :

Nuclear data represent numerical quantities that describe the properties of nuclei and the probabilities of their various physical interactions, such as :

- Decay data : radioactive constant, half-life, decay modes, spectrum of emitted radiation, …
- Nuclear reaction data : cross section for collision processes (for example between a neutron and a nucleus or between two nuclei), energy yields in the different reactions, …
- Atomic data : cross sections for collisions among electrons, atoms and molecules, ...
- Fission data : spectrum of fission product yields, number of emitted neutrons, fraction of delayed neutrons, …

2. Users of nuclear data :

Nuclear data are fundamental to the development and application of all nuclear sciences and technologies. They are essential ingredients in a wide range of applications, including the energy production, design of advanced reactors, management of nuclear waste, production and uses of radioisotopes, medical diagnostics and radiotherapy, applications of lasers and accelerators, fusion energy research, plasma processing, materials inspections and nuclear safeguards, geological and environmental work, fundamental science,..[1].

3. Nuclear data libraries :

These data (measured or evaluated) follow a complex process of evaluation, correction and analysis before being arranged in databases (libraries) which are directly employed in applications. The nuclear data libraries are classified into 3 categories:

- Bibliographic database : CINDA (the Computer Index of Neutron Data) is a database that contains bibliographic references to measurements, calculations, reviews and evaluations of neutron data. It also includes index references to computer libraries of neutron data available from four regional neutron data centers [2]. CINDA web database is https://www-nds.iaea.org/exfor/cinda.htm, see Figure I.1.

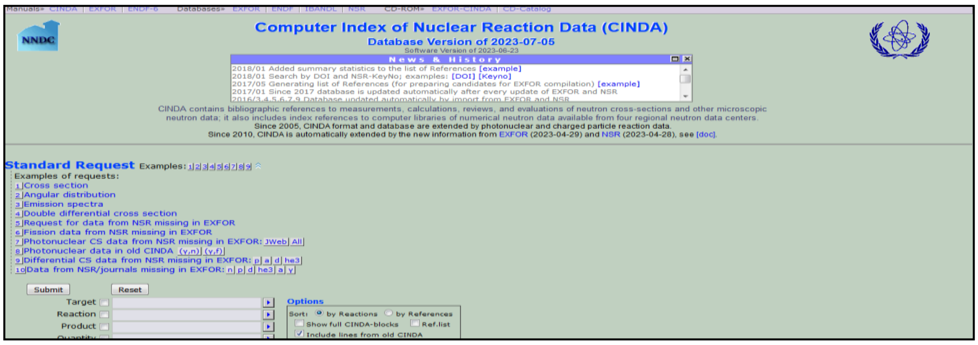


Figure.I.1. CINDA database.

- ***Experimental database :*** EXFOR (EXchange FORmat) contains an extensive compilation of experimental nuclear reaction data, comprising data from more than 22000 experiments [3]. EXFOR web database is https://www-nds.iaea.org/exfor/, see Figure I.2.

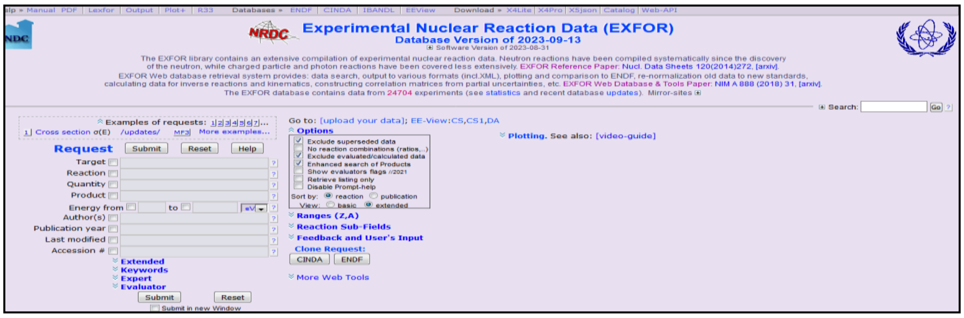


Figure.I.2. EXFOR database.

- ***Evaluated database :*** there are many regional and national evaluated data libraries : ENDF/B (USA) ; JEFF (NEA) ; CENDL (China) ; JENDL (Japan) ; BROND (Russia) … ; all in ENDF format [4]. ENDF web database is https://www-nds.iaea.org/exfor/endf.htm#1, see Figure I.3.

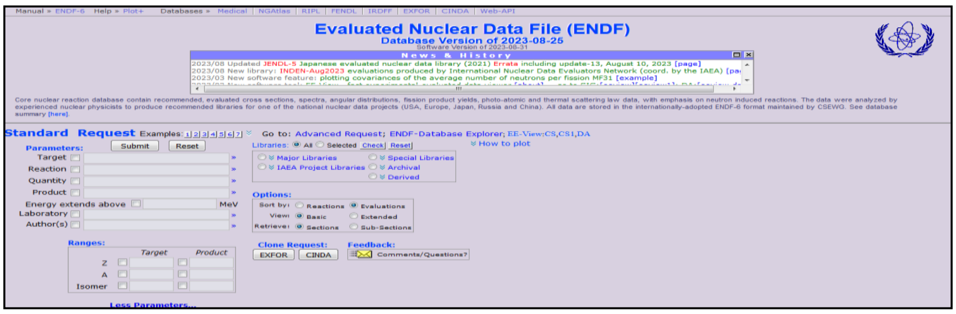


Figure.I.3. ENDF databases.

4. Evaluation of nuclear data :

The nuclear data must be "complete". Modern evaluations are done by combining the experimental data with nuclear model code calculations to extend or interpolate the available data [5].

4.1. Evaluation methodology :

The different steps to evaluate nuclear data are : measurement, compilation, evaluation, validation, adjustment and estimation of uncertainties in calculations [6].

The experimental nuclear data results are analyzed and compiled in computer form by one of the nuclear data centers and also enter the CINDA bibliographic index. The nuclear data centers spread across the world include [6,7] :

- The National Nuclear Data Center (NNDC) at Brookhaven National Laboratory in the USA: for North America.
- The Nuclear Energy Agency (NEA), located in Paris, France : for Western Europe and Japan.
- The Obninsk Center in Russia : for Eastern Europe.
- The Nuclear Data Services at the International Atomic Energy Agency (IAEA) in Vienna, Austria : for the rest of the world.

The IAEA coordinates the compilation activities of the four centers and collects all experimental data available in the EXFOR library [3,6].

For unmeasured or very complex data, the parameters of nuclear models are adjusted until the resulting data matches well with critical experiments. The results of evaluations are stored in Evaluated Nuclear Data Libraries which contain a combination of actual data and resonance parameters that can be reconstructed into pointwise data with specialized tools (such as NJOY or MCNP …). Figures I.4 and I.5 summarize evaluation and validation process, respectively [8]. Once the nuclear model code and ENDF formatting code used in the evaluation process is well verified and validated, the 4 boxes labeled Save contain the essential information that produced the nuclear data file. An input file for the nuclear model code with parameters adjusted to reproduce the available experimental data produces a complete set of nuclear reaction results. Finally, a formatting code produces the ENDF-6 data file, which is driven by a script that performs any additional actions. Validation with a

Monte Carlo approach, should give direct feedback to the data evaluation process and the evaluation and validation schemes become basically linked [8].

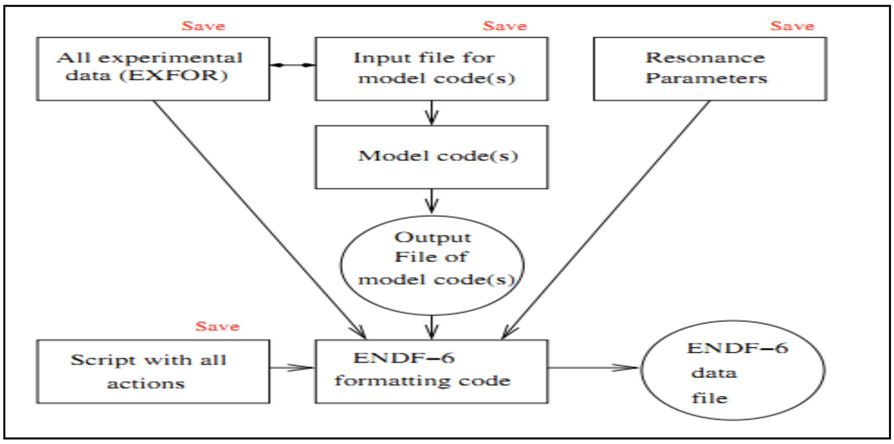


Figure.I.4. Flowchart of automated, reproducible evaluation process.

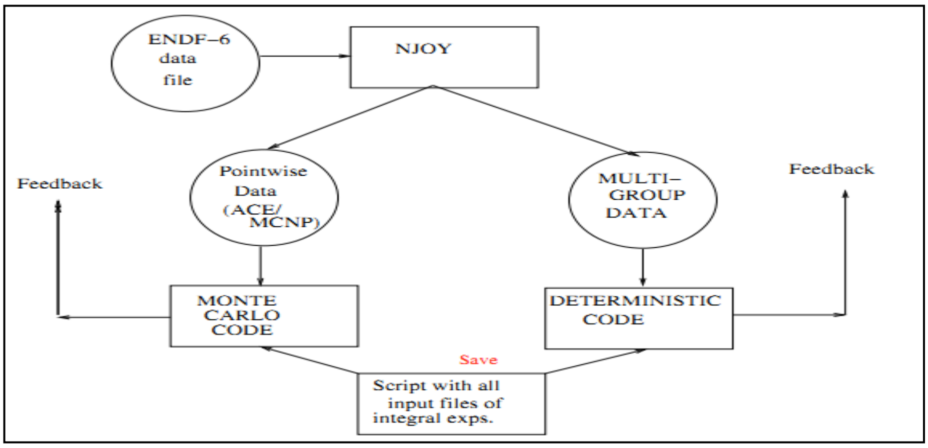


Figure.I.5. Flowchart of automated, reproducible validation process.

4.2. Evaluation projects :

The major evaluated nuclear data library projects are [6] :

- The American Evaluated Nuclear Data File (ENDF/B)
- The NEA Joint European Fission and Fusion File (JEFF)
- The Japanese Evaluated Nuclear Data Library (JENDL)
- The Russian Nuclear Data Library (BROND)
- The Chinese Evaluated Nuclear Data Library (CENDL)
- The Fusion Evaluated Nuclear Data Library (FENDL)

Chapter 2

Evaluated nuclear data libraries

1. Introduction :

All the evaluated nuclear data libraries are a collection of evaluated data stored in a defined computer readable format that can be used as the main input to nuclear data processing programs : the international ENDF format. The ENDF formats describe how the data are arranged in the libraries and give the formulas needed to reconstruct physical quantities from the parameters in the library. They were originally developed for use in the US national nuclear data files called ENDF/B [5,9].

2. Main libraries :

2.1. ENDF/B library (Evaluated Nuclear Data File) :

The ENDF formats and libraries are decided by the Cross Section Evaluation Working Group (CSEWG), a cooperative effort of national laboratories, industry, and universities in the USA and Canada, and are maintained by the National Nuclear Data Center (NNDC) at Brookhaven National Laboratory, New York. The ENDF files went through various versions. The formats were upgraded with each version to handle new features [5,9]. The historical releases of ENDF/B files are summarized below [10] :

- ENDF/B-I was released in July 1968 (ENDF-1 format)
- ENDF/B-II was released in August 1970 (ENDF-2 format)
- ENDF/B-III was released in late 1972 (ENDF-2 format)
- ENDF/B-IV was released in February 1975 (ENDF-4 format)
- ENDF/B-V : version V.0 was released in June 1979, V.1 in 1983, V.2 in January 1985 (ENDF-5 format)
- ENDF/B-VI : version VI.0 was released in July 1990, VI.1 in September 1991, VI.2 in June 1993, VI.3 in May 1995, VI.4 in December 1996, VI.5 over 1997-1998, VI.6 over 1998-1999, VI.7 over 1999-2000, and VI.8 in 2001 (ENDF-6 format)

- ENDF/B-VII : version VII.0 was released in December 2006, VII.1 in December 2011 (ENDF-6 format)
- ENDF/B-VIII : version VIII.0 was released in February 2018 (ENDF-6 format)

The first few versions (I, II, III) were largely intended for thermal reactor applications. ENDF/B-IV and ENDF/B-V shifted the emphasis toward fast reactor and fusion applications. ENDF/B-VI, ENDF/B-VII and ENDF/B-VIII have been made for charged particle and accelerator applications [5].

2.2. JEFF library (Joint Evaluated Fission and Fusion File) :

In Western Europe, there were two projects that were closely linked [6,11] :

- The Joint Evaluated File project, JEF, which was intended for fission reactors, and coordinated by the Nuclear Energy Agency`s Data Bank (NEA).
- The European Fusion File project, EFF, which was developed for fusion applications, and coordinated by the ECN (Energieonderzoek Centrum Nederland) of Petten in the Netherlands, Holland.

JEF and EFF are currently being combined within the framework of the JEFF project to produce the first version JEFF-3 library in 1999 [6,11]. The latest version JEFF-3.3 library has been officially released in November 2017 [12].

2.3. JENDL library (Japanese Evaluated Nuclear Data Library) :

This data file contains recommended nuclear data. The Nuclear Data Center (NDC) of Japan Atomic Energy Agency (JAEA) is making the JENDL library with the aid of Japanese Nuclear Data Committee (JNDC). The first version JENDL-1 library was released in 1977. The latest version JENDL-5 library released in 2021 [13].

2.4. BROND library (Biblioteka Rekomendovannykh Ocenennykh Nejtronnykh Dannykh) :

The computerized data library for evaluated neutron reaction data (BROND) was developed by the USSR Center, Obninsk, Russia. The first version BROND-NDS1 was released over 1987-1988 [14]. The new version BROND-3.1 library released in 2016 [15].

2.5. CENDL library (Chinese Evaluated Nuclear Data Library) :

The CENDL project has been initiated under the joint collaboration of CENDL working group, namely China Nuclear Data Center (CNDC) and the Chinese Nuclear Data Cooperation Network (CNDCN) since 1970s. The first version CENDL-1.0 library was released in 1985. The latest version CENDL-3.2 released in 2020 [16].

2.6. FENDL library (Fusion Evaluated Nuclear Data Library) :

The FENDL library was the response of the IAEA Nuclear Data Section to the need for a data library specifically designed for fusion applications. An initial meeting was held in 1989 and following the creation of FENDL-1 in 1995 [17]. The new version FENDL-3.2 released in 2021 [18].

Each of these libraries has adopted the ENDF-6 format as the common method for publishing their nuclear data files, but each has been able to define its own procedures for using the formats, reviewing the evaluations, and testing the results [5].

Chapter 3
ENDF Format

1. General description of the ENDF system :

ENDF formats were developed for the storage and retrieval of evaluated nuclear data to be used for applications of nuclear technology. Each ENDF evaluation has an hierarchical structure by a set of key parameters. Following is a list of these parameters and their definitions [5,9].

1.1. Library (NLIB, NVER, LREL NFOR) :

A library is a collection of material evaluations from a recognized evaluation group. Each of these collections is identified by NLIB number. The NVER, LREL and NFOR parameters defined the version, release and format of the library, respectively (see Tables III.1 and III.2) [9].

Key parameters		definition
Library	NLIB	A collection of specific evaluation (e.g. NLIB 0=ENDF/B)
Version	NVER	One of the periodic updates to a library in ENDF format (e.g. NVER=7 for ENDF/B-VII)
Release	LREL	Intermediate release containing minor updates and corrections of errors after a general release of a library (e.g. ENDF/B-VI.2 for release 2 of the ENDF/B-VI library)
Format	NFOR	Format in which the data is tabulated (e.g. NFOR=6 for ENDF-6)

Table.III.1. Key parameters defining the library.

NLIB	Library definition
0	ENDF/B - United States Evaluated Nuclear Data File
1	ENDF/A - United States Evaluated Nuclear Data File
2	JEFF – NEA Joint Evaluated Fission and Fusion File
3	EFF – European Fusion File
4	ENDF/B – High Energy File
5	CENDL – China Evaluated Nuclear Data Library
6	JENDL - Japan Evaluated Nuclear Data Library
33	FENDL – IAEA Fusion Evaluated Nuclear Data Library
35	BROND – Russian Evaluated Nuclear Data File (IAEA version)
41	BROND – Russian Evaluated Nuclear Data File (original version)

Table.III.2. Currently defined NLIB numbers.

1.2. Sub library (NSUB) :

The sub-library NSUB defines the incident particle (projectile) IPART and the type of data ITYPE :

NSUB = 10 * IPART + ITYPE

IPART = 1000 * Z + A ; where Z is the atomic number and A is the mass number for the particle ; use IPART = 0 for incident photons or no incident particle (decay data), use IPART=11 for incident electrons.

The sub-libraries allowed in ENDF-6 are listed in Table III.3 [9].

NSUB	IPART	ITYPE	Sub-library names
0	0	0	Photo-Nuclear Data
1	0	1	Photo-Induced Fission Product Yields
3	0	3	Photo-Atomic Interaction Data
4	0	4	Radioactive Decay Data
5	0	5	Spontaneous Fission Product Yields
10	1	0	Incident Neutron Data
11	1	1	Neutron Induced Fission Product Yields
12	1	2	Thermal neutron Scattering Data
113	11	3	Electron Atomic Interaction Data
10010	1001	0	Incident Proton Data
10011	1001	1	Proton Induced Fission Product Yields
10020	1002	0	Incident Deuteron Data
20040	2004	0	Incident Alpha Data

Table.III.3. Sub-library Numbers and Names.

Sub-libraries contain the data for different target materials identified by MAT numbers. Each material evaluation contains data blocks called Files, identified by the MF number. Sections within individual Files are identified by the MT numbers, which indicate the type of reaction [5,9].

1.3. Material (MAT) :

A material may be a single nuclide, a natural element containing several isotopes, or a mixture of several elements (compound, molecule, etc.). A single isotope can be in the ground state or an excited (isomeric) state. Each material in an ENDF library is assigned a unique identification number, designated by the symbol MAT, which ranges from 1 to 9999 (four digits) [9].

- The MAT numbers (Z01-Z99) have been allocated to each element Z, through Z = 98. Natural elements have MAT numbers Z00. The MAT numbers for isotopes of an element are assigned on the basis of increasing mass in steps of three, allowing for the ground state and two metastable states. The lightest stable isotope of an element has MAT number Z25.

- For the special cases of elements from einsteinium to lawrencium ($Z \geq 99$), MAT numbers 99xx are assigned, where xx = 30, 25, 20, 15, and 12 for elements 99 to 103, respectively.

- For compounds and molecules, MAT numbers between 0001 and 0099 are assigned on a special basis (Table III.4) [9].

Compound	MAT
Hydrogen	1-10
Deuterium	11-20
Lithium	21-25
Beryllium	26-30
Carbon	31-44
Oxygen	45-50
Metals	51-70
Fuels	71-99

Table.III.4. MAT numbers for compounds.

1.4. File (MF) :

A File is a block of data that describes a certain type of information. MF runs from 1 to 99 (two digits).The list of allowed Files (MF) and a description of their usage is given in Table III.5 [9].

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 section for photon production

Table.III.5. Definitions of File Types (MF).

1.5. Section (MT) :

Each section describes a particular reaction. MT runs from 1 to 999 (three digits) [9].

Total reaction : MT = 1

Elastic Scattering : MT = 2

Non-elastic Scattering : MT = 3

Radiative Capture : MT = 102

Simple Single Particle Reactions : reactions have only a single particle and a residual nucleus (and possibly photons) in the final state (Table III.6). For example, neutron emission MT= 50 leaves the residual nucleus in the ground state, MT= 51 leaves it in the first excited state, MT=52 in the second, and so on.

Discrete	Continuum	Discrete+Continuum	Emitted particle
50-90	91	4	n
600-648	649	103	p
650-698	699	104	d
700-748	749	105	t
750-798	799	106	^{3}He
800-848	849	107	α

Table.III.6. Definitions of MT numbers of simple single particle reactions.

Simple Multi-Particle Reactions : reactions have only two to four particles, a residual nucleus, and photons in the final state, and the residual nucleus does not break up. For example, MT=11 for 2nd ; MT=16 for 2n ; MT=17 for 3n ; MT=22 for nα ; MT=23 for n3α ; MT=25 for 3nα ; MT=28 for np ; MT=29 for n2α ; MT=30 for 2n2α ; ...

Break up Reactions : reactions can be described as proceeding in two steps : first, one or several particles are emitted as in the simple reactions described above, then the remaining nuclear system either breaks up or emits another particle. These reactions are represented using MT numbers (for the first step of reactions) and LR flags (for the second step of reactions) [10], see Table III.7.

Reaction	MT	LR
$^{12}C(n,n_2)^{12}C \rightarrow 3\alpha$	52	23
$^{6}Li(n,n_1)^{6}Li \rightarrow d+\alpha$	51	32
$^{7}Li(n,n_c)^{7}Li \rightarrow t+\alpha$	91	33
$^{10}B(n,n_{12})^{10}B \rightarrow d+2\alpha$	62	35
$^{16}O(n,n_1)^{16}O \rightarrow e^{+}+e^{-}+^{16}O$	51	40

Table.III.7. Some examples of TR values.

Fission : MT=18 for Total fission ; MT=452 for total number of neutrons per fission ; MT=455 for number of delayed neutrons per fission ; MT=456 for number of prompt neutrons per fission ; MT=458 for components of energy release in fission.

Other sections :

MF = 1 and MT = 451 : Heading or title information ; given in File 1 only.

MF = 2 and MT= 151: Resonance parameters ; File 2 contains only one section.

2. Representation of Data :

The data given in all sections always use the same set of units (see Table III.8) [9].

Quality	Units
Energies	Electron-Volts (eV)
Angles	Dimensionless cosines of angle
Cross sections	Barns
temperatures	Kelvin
Mass	Units of the neutron mass
Angular distributions	Probability per unit-cosine
Energy distributions	Probability per electron-Volt
Energy-angle distributions	Probability per unit-cosine per electron-Volt
Half life	Seconds

Table.III.8. Summary of ENDF units.

3. ZA, AWR, AWI and AWP numbers :

A Floating point number ZA is used to identify materials (and projectiles and reaction products). ZA is constructed by : ZA = (1000.0*Z) + A ; where Z is the atomic number and A is the mass number. For example, for isotope Uranium-235 ZA=92235.0 and MAT=9228

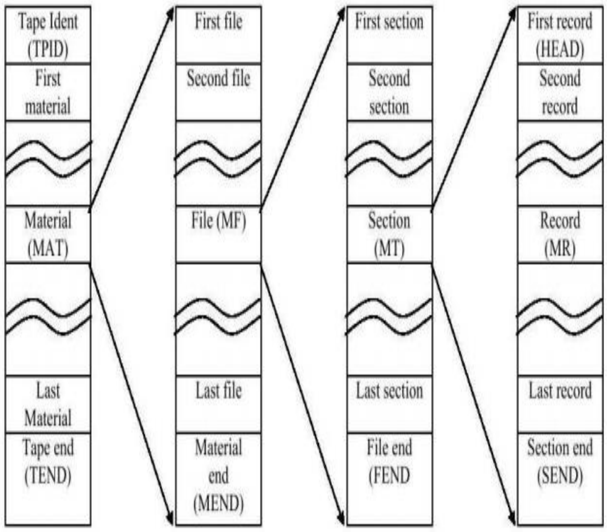
- If the material is a natural element, A is set to 0.0. For example, for the element tungsten ZA=74000.0 and MAT=7400.

- For compounds, ZA is calculated from ZA=MAT+100

A material, incident particle (projectile), or reaction product is also characterized by a quantity that is proportional to its mass relative to that of the neutron. Typically, these quantities are denoted as AWR, AWI, or AWP for a material, projectile, or product, respectively [9].

4. Structure of an ENDF Data Tape :

An ENDF tape is built up from a small number of basic structures called records. These records consist of one or more 80 character FORTRAN records. The tape contains the data of a material (identified by MAT), which is divided into files in increasing order by MF. Each file contains several sections in increasing order by MT. Finally, a section is divided into records. The beginning of a section is signaled by HEAD records. The end of a section, file, material or tape is signaled by SEND, FEND, MEND and TEND records, respectively (see Figure III.1) [5,9].




```
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
```

Figure.III.1. Structure of an ENDF data tape.

Practical Work 1
ENDF Libraries

1. Theoretical part : Discussion of some examples of nuclear data from ENDF library (ENDF/B-VII)

1.1. Cross section :

Simple cross section 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-VII :

```
 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
```

1.2. Resonance parameters :

The resonance parameter data are given in File 2 using MT=151. Here is an example of the resonance parameters for Pu-238 from ENDF/B-VII :

```
 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
 ...
```

1.3. Angular distributions :

The angular distributions for two body scattering reactions are given in either File 4 or a special “law” of File 6. The following is an example of a section of File 4 for elastic scattering for Cu-63 from ENDF/B-VII :

```
 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
```

1.4. Fission neutron yields :

The following example presents the total number of neutrons produced in the fission reaction of U-233 from ENDF/B-VII file :

```
 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
```

2. Practical part : Access different ENDF libraries (ENDF/B, ENDF/HE, BROND, CENDL, JENDL, FENDL, JEFF, IRDF, INDL, …) ; See the hierarchical structure of each ENDF library (Library, Sub-Library, ENDF Tape : MAT, MF, MT) ; Know to easily read an ENDF Data Tape from start (Tape Identification : MF = 1, MT = 451) to finish (TEND record : MAT = -1, MF = 0, MT = 0).

3. Answers (theoretical part) :

3.1. Cross section :

The following example presents the cross section for the (n,2n) reaction in natural silicon. The first line is the HEAD record ; it contains the ZA value (1000*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 (for example, at 14 MeV the cross section is 0.04663 barns). The last line in the section is the SEND record. Note that this is an endothermic reaction (negative Q value), and it has a threshold energy of 8.7781 MeV (calculated by using the formula –Q(AWR+1)/AWR).

3.2. Resonance parameters :

The following example presents the resonance parameters for Pu-238. The third line says the resonance range extends from 10^{-5} to 200 eV. The fourth line says that the target spin is 0 and the scattering length is 0.9309. This translates into a potential scattering cross section of $4\pi\alpha^2$ = 10.89 barns. The parameters for different resonances start on the sixth line : there is a negative energy resonance at -10 eV with J=1/2, a total width of 0.06217 eV, a scattering width of 0.01581 eV, a capture width of 0.045 eV and a fission width of 0.00136 eV.

3.3. Angular distributions :

The following example presents the angular distributions for elastic scattering for Cu-63. 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 10^{-5} eV. 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.

3.4. Fission neutron yields :

The following example presents the total neutrons per fission of U-233. The “2” in the fourth field on the first line says that the fission neutron yields are given in tabulated form. The tabulation given as energy, total neutrons per fission pairs with energies in eV (for example, at 10^{-5} eV the total number of neutrons per fission is 2.4947) starts on line 4.

Practical Work 2

PREPRO CODES

1. Theoretical part : Presentation of the ENDF/B Pre-processing codes (PREPRO)

1.1. Introduction :

The ENDF/B Pre-processing codes (PREPRO) are a collection of 18 module codes, which are designed to convert ENDF/B formatted evaluated data from the originally distributed form (ENDF format) to a form in which the data can be used in application codes. They run on any operating system (Windows, Linux, and MAC(OSX)). PREPRO output is compatible with FORTRAN, C and C++. These codes are developed and maintained by Dermott E. Cullen [19,20]. They have a long history extending back to the origins of the ENDF effort over 50 years ago. The last version (PREPRO 2023) is smaller and yet as fast and precise as preceding versions [20].

1.2. Brief description :

After you have installed the codes, execute ***VERIFY.BAT*** file to verify implementation. It will take between 5 minutes and an hour (depending on the speed of your computer), to run all the codes. The normal sequence in which the codes are used is described as like [20] :

ENDF2C : Convert ENDF data to FORTRAN, C and C++, compatible form.
LINEAR : Linearize cross sections.
RECENT : Reconstruct cross sections from resonance parameters.
SIGMA1 : Doppler broaden cross sections to any temperature.
ACTIVATE : Create activation cross sections MF=10 from cross sections MF=3 and multipliers MF=9.
LEGEND : Calculate/correct angular distributions.
SIXPAK : Convert double differential data (MF=6) to single differential.
SPECTRA : Linearize and tabulate neutron emission spectra (MF=5).
FIXUP : Correct format and cross sections, define cross sections by summation.

DICTIN : Update the section index in MF=1, MT=451.

MERGER : Retrieve and/or Merge evaluated data.

GROUPIE : Calculate multigroup cross sections and multiband parameters.

COMPLOT : Plot a comparison of cross sections from two different evaluations.

EVALPLOT : Plot evaluated data.

MIXER : Calculate the cross sections for a combination of materials (mixtures).

VIRGIN : Perform exact uncollided (virgin) transmission calculations.

RELABEL : Relabel and sequence programs.

CONVERT : Convert codes for computer/precision/compiler.

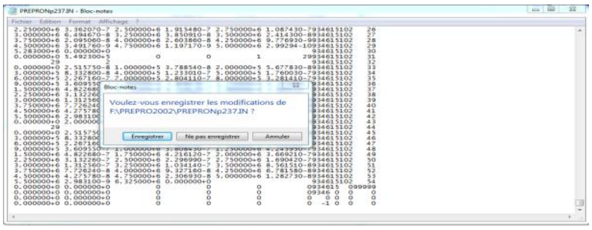
1.3. Use of the codes in combination :

In order to run any number of these codes in combination, one after the other, all you need is the facility to :

1) start a program, 2) rename a file, 3) delete a file, if you want to minimize disk space.

For example, if I want to run the sequence of codes, ENDF2C, LINEAR, RECENT, SIGMA1, ACTIVATE, LEGEND, FIXUP, DICTIN and EVALPLOT, I follow the following steps :

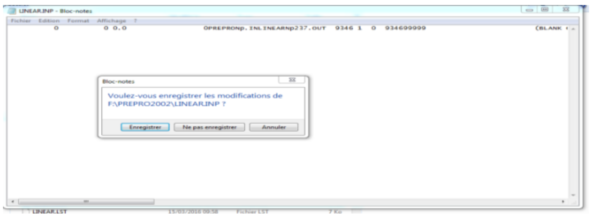
ENDF2C.IN



LINEAR.IN

rename ENDFB.INLINEAR.OUT

LINEAR

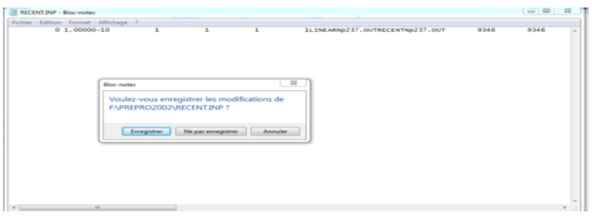


RECENT.IN

rename LINEAR.OUTRECENT.OUT

RECENT

delete **LINEAR.OUT**

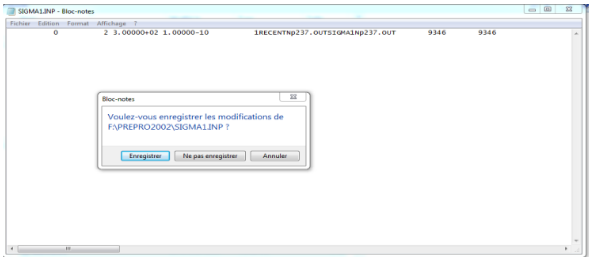


SIGMA1.IN

rename RECENT.OUTSIGMA1.OUT

SIGMA1

delete **RECENT.OUT**

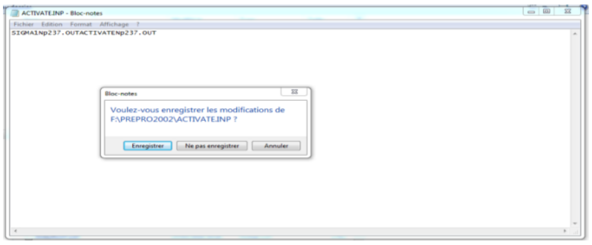


ACTIVATE.INP

rename SIGMA1.OUTACTIVATE.OUT

ACTIVATE

delete **SIGMA1.OUT**

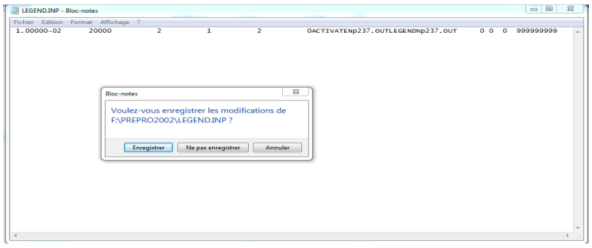


LEGEND.INP

rename ACTIVATE.OUTLEGEND.OUT

LEGEND

delete **ACTIVATE.OUT**

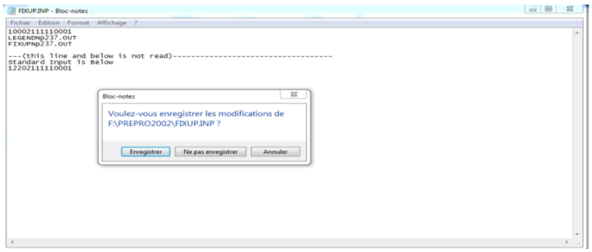


FIXUP.INT

rename LEGEND.OUTFIXUP.OUT

FIXUP

delete **LEGEND.OUT**

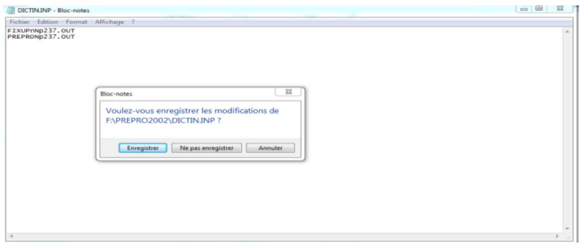


DICTIN.INP

rename FIXUP.OUTPREPRO.OUT

DICTIN

delete **FIXUP.OUT**

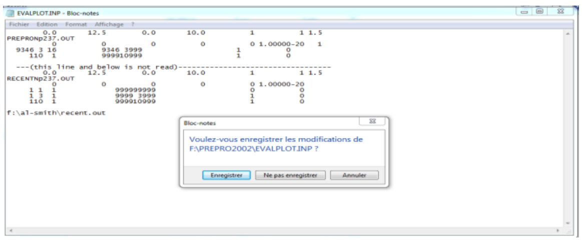


EVALPLOT.INT

rename PREPRO.OUT

EVALPLOT

delete **DICTIN.OUT**



Note, when each code finishes, renames the data output by the code to the filename of the data input to the next code. When the next code finishes, the data input to it is deleted (we no longer need it), and the cycle starts for the next code. More efficiently you could have defined ENDF input and output file names in the input parameter files for each code to link them together [20].

2. Practical part : Running PREPRO codes to plot fission (MT=18) and radiative capture (MT=102) cross sections (MF=3) for Am-241 (MAT=9543) from ENDF/B-VII.0

ENDF2C :

Copy Am-241 Tape (MAT=9543) from ENDF/B-VII.0

LINEAR :

```
0          0 0.0              0                    PREPRO.INLINEARAm241.OUT     9543 1  0  9543
3999
```

RECENT :

```
0 1.00000-10         1         1         1         1      LINEARAm241.OUTRECENTAm241.OUT
9543     9543
```

SIGMA1 :

```
0         2 0.00000+02 1.00000-10         1       RECENTAm241.OUTSIGMA1Am241.OUT
9543     9543
```

ACTIVATE :

```
SIGMA1Am241.OUTACTIVATEAm241.OUT
```

LEGEND :

```
1.00000-02   20000      2      1      2      0ACTIVATEAm241.OUTLEGENDAm241.OUT     0
0 0 999999999
```

FIXUP :

```
10002111110001
LEGENDAm241.OUT
FIXUPAm241.OUT
```

DICTIN :

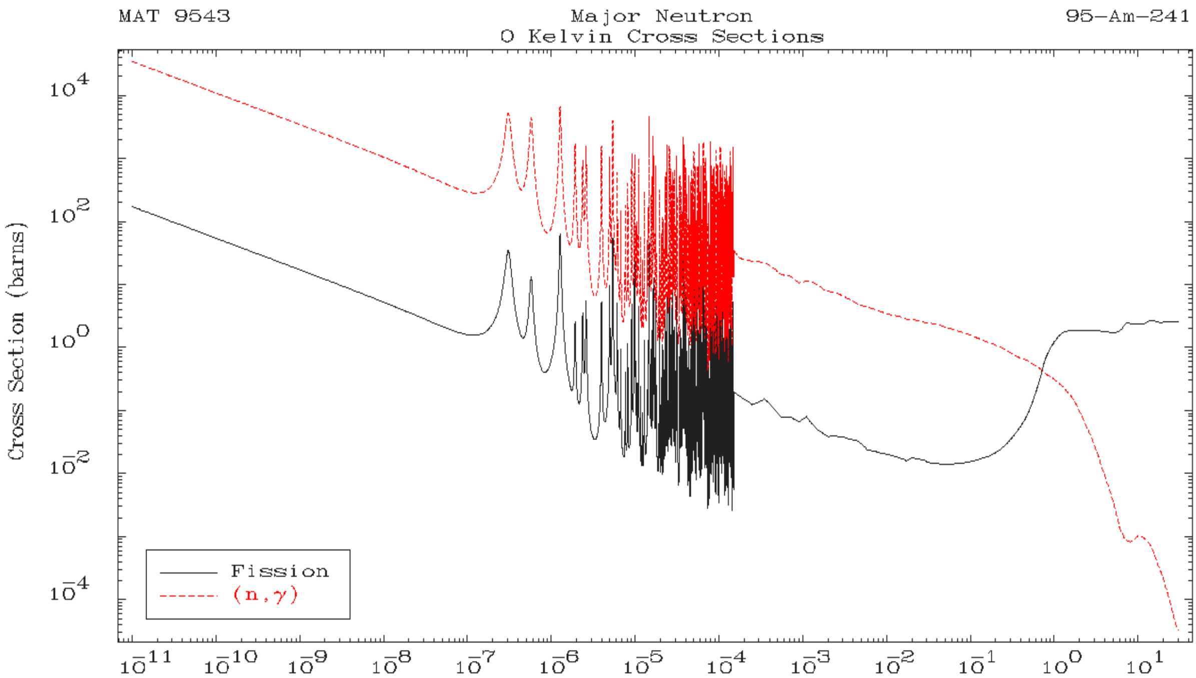
FIXUPAm241.OUT
PREPROAm241.OUT

EVALPLOT :

```
0.0      12.5      0.0      10.0        1         1 1.5
PREPROAm241.OUT
        0         0         0         0        0 1.00000-20   1
 9543 3 18           9543 3102                 1          0
   110  1          999910999                  1          0
```

3. Answers (practical part) :

Running PREPRO codes to plot fission (MT=18) and radiative capture (MT=102) cross sections (MF=3) for Am-241 (MAT=9543) from ENDF/B-VII.0 :



Incident energy MeV

Practical Work 3

JANIS CODE

1. Theoretical part : Presentation of the Java-based Nuclear Information Software (JANIS)

1.1. Introduction :

Java-based Nuclear Information Software (JANIS) is a display program designed to facilitate the visualization and manipulation of nuclear data (experimental data (EXFOR), bibliographical data (CINDA), evaluated data (ENDF/B, BROND, CENDL, FENDL, JENDL, JEF, ...)). It offers maximum flexibility for the comparison between these different nuclear data sets [21].

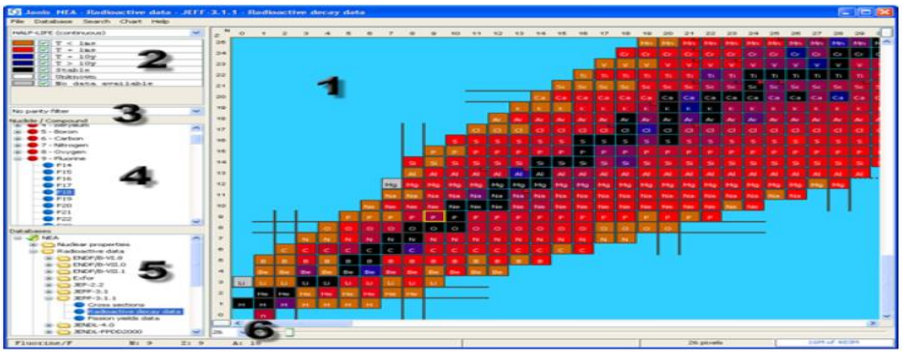
JANIS is the successor to JEF-PC, a software developed by the OECD Nuclear Energy Agency, the CSNSM-Orsay and the University of Birmingham. As JANIS is written in the Java language, it runs on almost all computer operating systems (Windows, Linux, UNIX, ...). The first version (JANIS 1.0) was released in October 2001 [21], and the latest version (JANIS 4.1) released in September 2020 [22].

1.2. General description :

Once Java has been installed, JANIS can be run directly from the DVD or installed on the computer (copy of the DVD content to the hard drive). To start JANIS you double-click on ***janis.bat*** file (Windows).

1.2.1. The Browser window :

When JANIS is started the main window named ***Browser*** displays the components shown in Figure 3.1 [21].

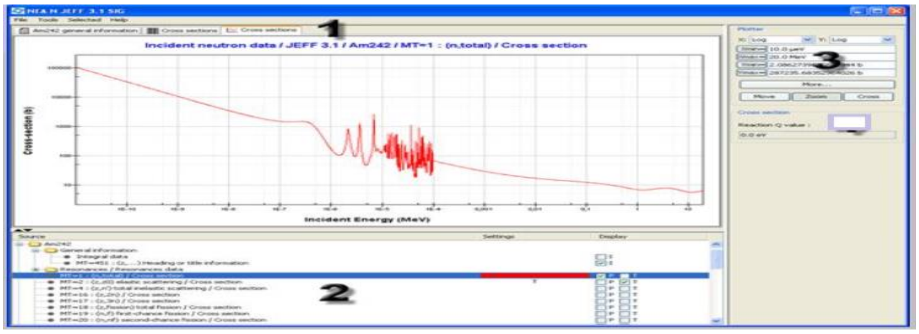


1. Chart of Nuclides
2. Legend for the Chart of Nuclides
3. Parity filter
4. Nuclide Explorer
5. Database Tree
6. Zoom controls

Figure.3.1. Browser

1.2.2. The Renderer window :

The ***Renderer*** window is the major window for data displaying in JANIS (see Figure 3.2). You can open more than one ***Renderer*** window at a given time [21].



1. Display panels
2. Selection Tree
3. Display panel parameters

Figure.3.2. Renderer

1.2.3. Plot views :

JANIS can display several kinds of plots : continuous plots, scatter plots with error bars, rays plots, and colormap plots (Figure 3.3) [21].

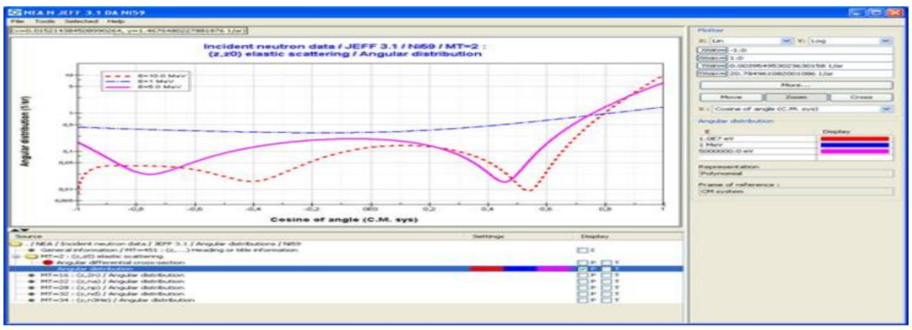


Figure.3.3. Plot views

1.2.4. Table views :

Table views display the title in the upper row followed by the data in a table made of header cells and data values in following cells (Figure 3.4) [21].

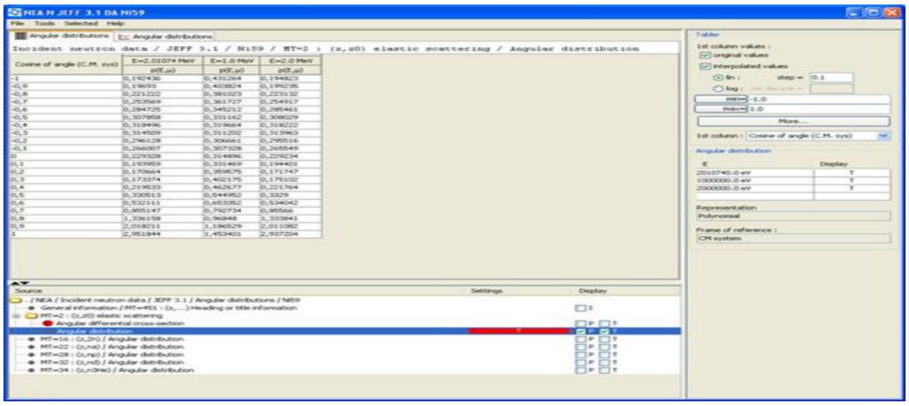


Figure.3.4. Table views

1.2.5. Text views :

JANIS displays several kinds of data as text panes (for example ENDF: "General information" which contains a summary description of the evaluation work (source of data, analysis method…) and a dictionary of available files and reactions, this is taken directly from (MF=1, MT=451)) [21], see Figure 3.5.

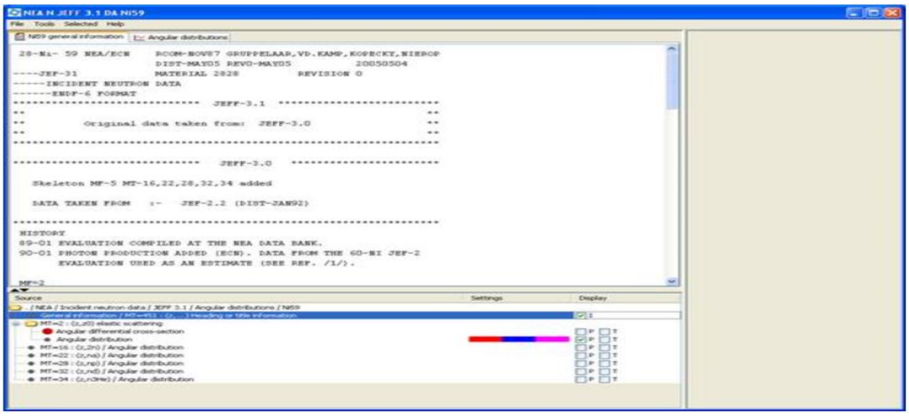


Figure.3.5. Text view

1.2.6. Decay path views :

Decay information is available under the "Radioactive data" node in the "Database Tree" [21] (Figure 3.6).

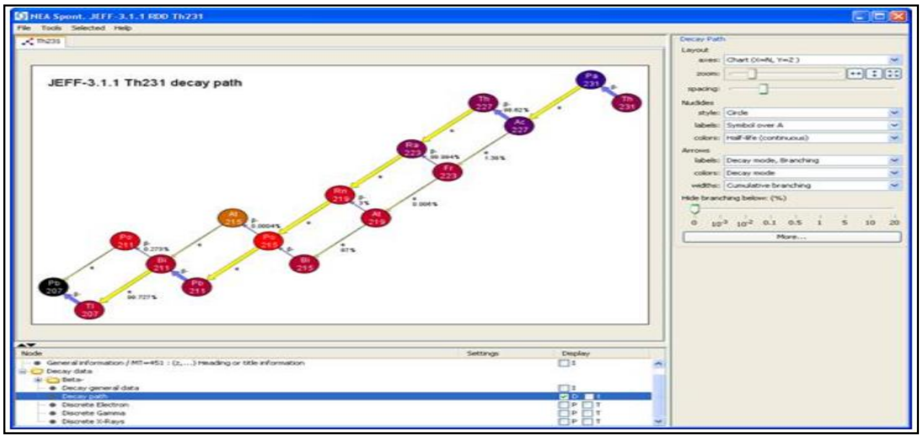


Figure.3.6. Decay path

1.3. Accessible data through JANIS :

A. Cross-sections and resonance parameters : Both pointwise and groupwise cross-section data can be displayed and compared. Resonance data can also be plotted to show $\sigma_0(E0)$ as a function of E_0 [21].

C. Energy distributions : The energy distribution gives the probability of emission of a secondary particle at a given energy E′. This probability also depends on the energy E of the incident particle, and it is generally represented as $p(E \rightarrow E')$ [21].

D. Angular distributions : angular distributions of secondary particles are given in MF=4 of the ENDF format as f(µ,E), where µ is the cosine of the angle between the incident and emergent particles and E is the energy of the incident particle. It may be defined in either centre-of-mass or laboratory systems [21].

E. Energy-angle distributions : The distribution in energy and angle of the reaction products is described in File 6 of the ENDF format. It provides an alternative and more accurate representation of the reaction products' characteristics compared to the separate representations using energy distribution (File 5) and angular distribution (File 4) [21].

F. Decay data : Decay data can be obtained from the "Radioactive decay data" node. Under this node, the "Chart of Nuclides" will display the mass of the nuclide, its excitation energy, the spin and parity, the half-life, the mean decay energies and decay modes [21].

G. Fission yields : Fission yield data depend on the projectile causing the fission (e.g. neutron-induced fission), its energy and the fissioning system. Fission may also occur as a radioactive process without projectile. Consequently, the “Fission yields data” node in JANIS might appear under several categories: “Radioactive data” (for spontaneous fission yields) or “Incident neutron data” [21].

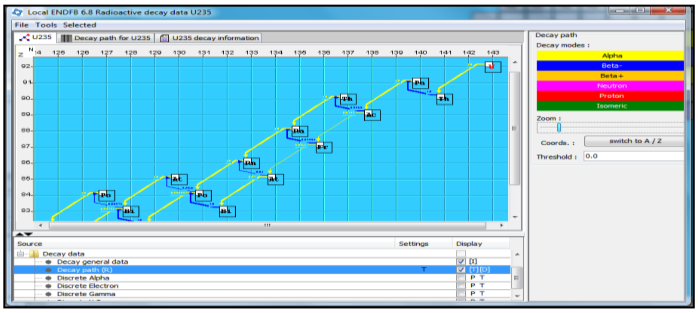
H. Photon production data : Photon production data are stored in ENDF File 12-15. They can be : multiplicities and transition probabilities arrays ; photon production cross sections ; photon angular distributions ; and continuous photon energy spectra [21].

2. Practical part : Running JANIS code to visualize and manipulate the nuclear data (radioactive data, interaction data (incident neutron (cross-sections, resonance parameters, energy and angular distributions, fission yields, …)., incident proton, incident alpha, incident gamma, incident electron, …), nuclear properties, from different databases (EXFOR, ENDF/B JEF, JEFF, JENDL, …).

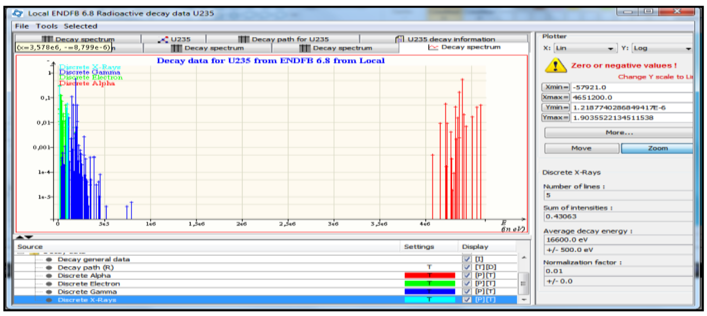
3. Answers (practical part) :

Running JANIS code to visualize and manipulate some nuclear data :

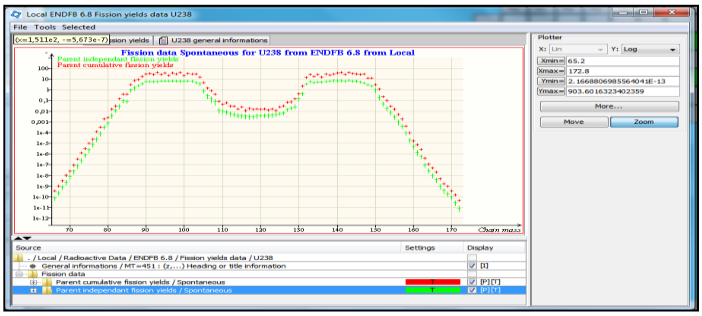
- Decay path for U-235 from ENDF/B-VI.8 library :



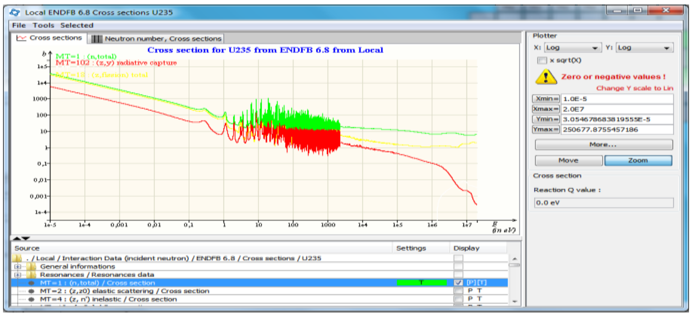
- Spectrum of emitted particle for U-235 :



- Spectrum of fission product yields for U-238 :



- Reaction cross section for U-235 :



- Other data ….

REFERENCES :

[1] : Nuclear Data Services ; IAEA ; 2020 ; https://www.iaea.org/resources/databases/nuclear-data-services

[2] : Computer Index of Neutron Data CINDA ; IAEA ; https://www.iaea.org/resources/databases/computer-index-of-neutron-data

[3] : Experimental Nuclear Reaction Data EXFOR ; IAEA ; https://www.iaea.org/resources/databases/experimental-nuclear-reaction-data

[4] : Evaluated Nuclear Data File ; IAEA ; https://www.iaea.org/resources/databases/evaluated-nuclear-data-file

[5] : R.E. MacFarlane ; An Introduction to the ENDF Formats ; Los Alamos National Laboratory ; USA 2000 ; https://inis.iaea.org/collection/NCLCollectionStore/_Public/38/027/38027897.pdf?r=1

[6] : Les données nucléaires ; Situation et perspectives ; Nuclear data ; Situation and future projects ; Direction des Etudes et Recherche ; EDF ; 1995 ; https://inis.iaea.org/collection/NCLCollectionStore/_Public/27/019/27019541.pdf?r=1&r=1

[7] : Patrick Talou ; Nuclear Reactors, Evaluations, Library Development ; Los Alamos National Laboratory; USA 2014 ; https://ejc2014.sciencesconf.org/conference/ejc2014/pages/talou_ejc2014-2.pdf

[8] : A.J. Koning ; New working methods for nuclear data evaluation : how to make a nuclear data library? ; International Conference on Nuclear Data for Science and Technology 2007 ; Article available at http://nd2007.edpsciences.org or http://dx.doi.org/10.1051/ndata:07683

[9] : A. Trkov & al ; ENDF-6 Formats Manual ; Report BNL-203218-2018-INRE ; 2021 ; https://www.nndc.bnl.gov/endf/format/endf-manual-latest.pdf

[10] : Evaluated nuclear data file (ENDF) ; ENDF/B Releases ; https://www.nndc.bnl.gov/endf/extended.html

[11] : Evaluations in JEF-2.2 ; Part 1 ; https://www.oecd-nea.org/dbdata/nds_jefreports/jefreport-17/chapter1.pdf

[12] : Joint Evaluated Fission and Fusion (JEFF) Nuclear Data Library ; JEFF library releases ; JEFF-3.3 ; https://www.oecd-nea.org/dbdata/jeff/

[13] : Japanese Evaluated Nuclear Data Library, JENDL ; Nuclear Data Center ; Japan Atomic Energy Agency ; https://wwwndc.jaea.go.jp/jendl/jendl.html

[14] : V.N. Manokhin & al ; BROND : USSR Evaluated Neutron Data Library ; IAEA-NDS ; 1989 ; https://inis.iaea.org/collection/NCLCollectionStore/_Public/21/077/21077861.pdf?r=1

[15] : BROND-3.1 ; NEA ; https://www.oecd-nea.org/jcms/pl_20505/evaluated-nuclear-data-library-descriptions#BROND31

[16] : Zhigang Ge & al ; CENDL-3.2 : The new version of Chinese general purpose evaluated nuclear data library ; 2020 ; https://doi.org/10.1051/epjconf/202023909001

[17] R.A. Forrest & al ; FENDL-3 Library ; Summary documentation IAEA ; INDC(NDS)-0628 ; December 2012 ; http://nds.iaea.org/publications/indc/indc-nds-0628.pdf

[18] Davide Laghi & al ; Application of JADE V&V capabilities to the new FENDL v3.2 beta release ; IAEA 2021 ; https://iopscience.iop.org/article/10.1088/1741-432/ac121a

[19] ENDF Pre-processing Codes ; IAEA ; https://www.iaea.org/resources/databases/endf-pre-processing-codes

[20] D.E. Cullen ; "PREPRO 2023 : ENDF/B Pre-processing Codes" ; Report IAEA-NDS-0241 ; June 2023; http://redcullen1.net/HOMEPAGE.NEW/Papers/PREPRO2023/PREPRO2023.pdf

[21] JANIS 4.0 USER'S GUIDE ; SEPTEMBRE 2013 ; https://oecd-nea.org/janis/janis-4.0/documentation/janis-4.0_manual_rev1.pdf

[22] JANIS Changes ; Latest version : JANIS 4.1 ; September 2020 ; NEA ; https://oecd-nea.org/jcms/pl_39929/janis-changes