Manual of the MOMENTOF code

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1 Introduction

The MOMENTOF code was originally developed to generate probability tables for reactor physics calculations, but it has several convenient functionalities to manipulate ENDF-formatted file, so some of them are explaned in this text.

2 How to install

Please get a package of the MOMENTOF code from the web site and freeze the tgz file.

The MOMENTOF code is written in the FORTRAN language, so one has to compile it to get the load module. How to compile is described in the ReadMe file in the directory 'momentof'. If the compilation is successfully done, the load module 'momentof.lm' will be generated in the 'momentof' directory, so please conform it. Compiler of gfortran is required, so if it is not yet installed in your computer, please install it first.

3 Reading point-wise cross section data files

The MOMENTOF code can read point-wise cross section data from the point-wise evaluated nuclear data files (PENDF) [1] and write external file in a simple format. You have a PENDF file of plutonium-241 in the directory '/momentof/PENDF'. This is originated from a plutonium-241 file in JENDL-4.0.

Please move to the directory 'Sample', and run a shell script 'go.pdf.sh'. If you do so, you can see a new file 'output_pdf' in the same directory. In this file, a pair of neutron incident energy and cross section value at the energy is given. With this text data, you can easily plot the cross section data.

This example gives the elastic scattering cross section data. If you want to do the same manipulation on different type of nuclear data, you can modify the input file 'input_pdf'. The forth line of this file defines the reaction type by the MT number. The MT number and its correspondence is given in **Table 1**.

When you want to process different nuclear data file, you can change the first line of the shell script 'go.pdf.sh'.

[1] PENDF can be generated from original ENDF files by some nuclear data file processing codes such as NJOY and PREPRO.

Table 1: MT number and its corresponding reaction type in the ENDF format

MT	Reaction type
1	Total
2	Elastic scattering
4	Inelasctic scattering
16	(n,2n)
18	Fission
102	Capture
452	The number of neutrons by fission reaction
451	η-value ($\nu\sigma_f/\sigma_a$)
42	P1 Legendre coefficient of elastic scattering

4 Calculations of multi-group infinite-dilution cross section

The MOMENTOF code has a capability or calculating multi-group cross section data from PENDF.

Multi-group infinite-dilution cross section is calculated with the following equation:

$$\sigma_g = \frac{\int_{E\in g} \sigma(E)W(E)dE}{\int_{E\in g} W(E)dE}, \tag{1}$$

where $W(E)$ is a weight function and it can be arbitrarily chosen from several candidates by users in MOMENTOF.

Let us try to run the shell script 'go.inf.sh' in the directory 'Sample'. If you do so, you can get a file 'output_inf' in which 107-group infinite-dilution cross section of plutonium-241 is given. In addition, simple text data is given in the file 'siginf'.

If you want to calculate multi-group cross sections of different nucides, you can change the PENDF file in the first line of 'go.inf.sh'. If you want to calculate multi-group cross sections with different group structure, you can modify the third line of the 'input_inf' file. In the case of SRAC 107-group, a pair of 107 and 4 should be used, but in the case of JFS 70-group, a pair of 70 and 1, or in the case of VITAMIN-J 175-group, a pair of 175 and 2 shoule be used.