Technical Material

Development and verification of a new nuclear data processing system FRENDY

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JAEA developed the new nuclear data processing system FRENDY in order to solve the problems of the current widely used nuclear data processing systems and process the new evaluated nuclear data file. Verification of FRENDY was carried out by three steps, *i.e.*, verification of each function, comparison of the results, and comparison of the k_{eff} values for the 79 benchmark experiments in the ICSBEP handbook using cross section data library processed by FRENDY with those by NJOY99. These results verified that FRENDY generates the ACE file correctly.

Keywords; FRENDY; JENDL; nuclear data processing; cross section; code library; ICSBEP; benchmark; MCNP

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1. Introduction

Japan Atomic Energy Agency (JAEA) has produced an evaluated nuclear data file JENDL [1] and several nuclear analysis codes, *e.g.*, a versatile reactor analysis code system MARBLE2 [2], a comprehensive neutronics calculation code system SRAC [3-4], a general purpose Monte Carlo code MVP [5], and a particle and heavy-ion transport code system PHITS [6]. These codes require cross section data libraries which are generated from evaluated nuclear data files such as JENDL, ENDF [7], and JEFF [8]. The cross section is one of the most fundamental quantities for these codes. The evaluated nuclear data file has been updated over the years with reflecting progresses on the nuclear modeling, new experimental data and a number of feedbacks from the benchmark analysis. The cross section data library of neutronics calculation codes is generated by processing the evaluated nuclear data file when it is updated. Though the nuclear data processing is very important process connecting the evaluated nuclear data file and the nuclear analysis codes, the nuclear data processing system had not been developed in Japan [9].

Currently, NJOY [10-11] of Los Alamos National Laboratory (LANL) and PREPRO [12] of International Atomic Energy Agency (IAEA) are widely used for that process in the world and these codes are also used for the generation of cross section data libraries of JAEA's application codes. NJOY was an exclusive nuclear data processing system that generates A Compact ENDF (ACE) file which is used for the continuous energy Monte Carlo code MCNP [13], PHITS and so on, for several decades. The other nuclear data processing systems, *e.g.*, PREPRO, GAIA [14], GALILEE [15], and Ruller [16] use NJOY modules to generate the ACE file.

For the nuclear data processing, we have been required to overcome the following problems:

1) Incomplete nuclear data processing for a new version of JENDL,

2) Generation of the cross section data library of neutronics calculation codes by end users,

3) Adoption of a new nuclear data format.

If the current nuclear data processing systems perfectly process the evaluated nuclear data based on our experience, there is no problem with using these systems. However, when a new version of JENDL was released, we usually found serious problems of these systems and needed to revise them. Furthermore, if development or distribution of these systems is stopped or critical bugs are found, that may cause a large impact on the current JENDL dissemination as well as the future of the calculation code development in Japan. We are also worried about unexpected stop of distribution, because of restriction and/or export control by the other governments.

Another problem is related to the cross section generation by end users. JAEA has no right to release the current nuclear data processing system which uses modules of NJOY and PREPRO. When end users want to use the new evaluated nuclear data file, they have to wait for revisions of NJOY and PREPRO to appropriately process it and release of the new cross section data library processed by JAEA.

Recently, changing the nuclear data format is considered by adopting the Generalized Nuclear Data (GND) format [17] by utilizing the Extensible Markup Language (XML). Since the GND format is completely different from the current ENDF-6 format [18] which was defined several decades ago, the current widely used nuclear data processing systems, *e.g.*, NJOY and PREPRO, cannot treat such a new format without extensive modification. To process the evaluated nuclear data written in the new format, development of the new nuclear data processing system is desired. To handle the GND format, several nuclear data processing systems, *e.g.*, FUDGE [19], NJOY21 [20], AMPX-2000[21], GAIA, GALILEE, and Ruller are under development in a few countries.

To overcome the problems mentioned above, JAEA started to develop the new nuclear data processing system FRENDY (FRom Evaluated Nuclear Data librarY to any application) in 2013. FRENDY is developed in order to solve the problems of the current widely used nuclear data processing systems and immediately process the new evaluated nuclear data file. Since all of other nuclear data processing systems use NJOY modules to generate the ACE file, FRENDY is the world's first nuclear data processing system to generate the ACE file without NJOY modules. When a new evaluated nuclear data file is released and this evaluated nuclear data file cannot be appropriately processed by the current processing systems, we will be able to easily identify the cause of the problem by comparing the results of the FRENDY with those of the current ones.

In this paper, the outline of the development, capabilities, and performances of FRENDY are described. This paper is composed of five sections. In Sec. 2, an overview of the FRENDY is explained. In Sec. 3, the verification of each module of FRENDY is investigated. To verify FRENDY, the results of FRENDY are compared with those of NJOY99 (updates 393) [10]. The processing time is also estimated in Sec. 3. In Sec. 4, the k_{eff} values of MCNP5 using the ACE files processed by FRENDY are compared with those by NJOY99 in order to verify the ACE file generated by FRENDY. A total of 79 benchmark experiments, representing 752 critical configurations in the ICSBEP handbook [22] are used for the comparison. Finally, concluding remarks and future works are provided in Sec. 5.

2. Overview of FRENDY

FRENDY is developed to solve the problems of the current nuclear data processing systems such as NJOY and PREPRO. The features of FRENDY are as follows:

1) Utilization of modern programming techniques,

2) Consideration of maintainability, modularity, portability, and flexibility,

3) Applicable to the new nuclear data format.

To improve maintainability and modularity, it was written in the object-oriented language C++. Since all classes were encapsulated, the modification of the class does not affect the other classes, and each class can be easily reused in other codes. Furthermore, to enhance portability and flexibility, each class is carefully designed to minimize the function and to maintain the independence of each module. To assure quality and credibility, each function is also well verified using the "Boost.Test" library which is one of the unit testing frameworks [23]. For the source code management, the version control system Git [24] was introduced.

The schematic diagram of FRENDY structure is shown in **Figure 1**. The solid lined modules have been already implemented, while the dot-lined ones have not been developed yet because the GND format is not finalized. Each module can be easily revised and extended for the future needs and imported to the other calculation codes by adding only a few lines without deeply understanding of them. For example, if developers want to implement the Doppler broadening module in their code, they can use it when the "NuclearDataObject" class and the "DopplerBroader" module are imported into their code. In the future, when the nuclear data processing system is not required, the nuclear analysis codes will directly read and process the evaluated nuclear data file in these codes by implementing the modules of FRENDY.

As shown in Figure 1, FRENDY converts a set of evaluated nuclear data in the ENDF-6 format to an object of the "NuclearDataObject" class which has its own data structure. Using the "NuclerDataObject" class, FRENDY can be easily extended to support a new nuclear data format. Though the current version of FRENDY supports the traditional ENDF-6 format only, it becomes possible to address the GND and other nuclear data formats if new parser, writer and converter classes are implemented.

Verification of evaluated nuclear data files is also considered in this system. When evaluated nuclear data is read, FRENDY automatically checks the data format, *e.g.*, difference of mass number in each MF/MT number, the interpolation type, the order of the table data, and so on. The checking function will be improved in the future for the verification of next generation of JENDL.

< Figure 1 >

To develop FRENDY by gathering the appropriate knowledge, a development team was founded in Nuclear Science and Engineering Center of JAEA [25], consisting of the nuclear data evaluators and specialists on the neutronics calculations. Development of this system is also supported by the nuclear data processing working group in the JENDL committee that was organized for the promotion of nuclear data research. The development scheme is shown in **Figure 2**. In order to obtain the needs and feedback from potential users in Japan, the development status has been reported in this working group. The development team reflects their useful comments and requirements on the development and the improvement of FRENDY.

< Figure 2 >

3. Verification of FRENDY

Verification of FRENDY system is carried out in the following processes:

1) Verification of each function,
2) Comparison of the results, *e.g.*, cross section and angular distribution to verify each module,
3) Comparison of the k_{eff} values of the integral experiments using cross section data processed by FRENDY with those by other processing systems to verify combination of each module.

The verification of each function is required to ensure that it works as intended. It is carried out in order to find bugs in the code by using the "Boost.Test" library. The comparison of the results is to verify each module and to confirm that the calculation method and calculation flow are appropriate. In this verification, we investigate the cause of the difference of the results, if any. Since the differences of the results affect the neutronics calculation, it is important to analyze the results of the neutronics calculation. The comparison of the results is described in this section and that of the neutronics calculation results is given in Sec. 4.

To verify each module, the results are compared with those of NJOY99 (updates 393) [10]. All nuclei and materials prepared in JENDL-4.0 [1] are used for the verification. Since NJOY99 is widely used in the world, we compare the FRENDY results with the NJOY99 ones. The current version of FRENDY also adopts similar data processing methods implemented in NJOY99 [26] as follows:

1) ψ-χ method is used to calculate Single-Level Breit-Wigner [26],
2) Kernel broadening method is used for the Doppler broadening [27],
3) Ladder approach is used for the calculation of probability table [28].

The fractional tolerance used for linearizing in resonance reconstruction, Doppler broadening and processing thermal scattering law data is 0.01%. Generally, the tolerance value is 0.1% to generate ACE files. In this study, to compare the cross section values which are obtained by FRENDY with those by NJOY99 in more detail, such a strict tolerance value was used.

3.1. Verification of Resonance Reconstruction and Doppler Broadening

Comparisons are made for the total, elastic scattering, fission, and radiative capture cross sections of ^{238}U from JENDL-4.0. **Figures 3 to 6** show the Doppler broadened cross sections at 296 K and the relative difference. The differences of the cross section in Figures 3 to 6 include differences of both resonance reconstruction and Doppler broadening since resonance reconstruction is required to calculate the Doppler broadened cross sections.

As shown in Figures 3 to 6, the large difference is observed at 2×10^4 eV in all reactions. This energy grid is the boundary of the resolved and unresolved resonance regions and the cross section is discontinuity at this energy grid. Since the calculation method of the cross section at the boundary is different, this large difference is observed. NJOY99 considers that the cross section at the boundary is the average of the resolved and unresolved resonance cross section. FRENDY considers that the cross section at the boundary is the resolved resonance cross section. Since the relative difference of the boundary and neighbor energy grid is less than 1.0×10^{-6}, this difference has no impact on the neutronics calculation.

The cause of the other difference is the energy grid number. **Figure 7** shows the detail of the difference region of the fission cross section. The energy grid number of FRENDY is larger than that of NJOY99. Though the calculation results of FRENDY are written in the PENDF file, the format of numerical data is different. FRENDY uses the decimal notation instead of the scientific notation which is used in NJOY99, when data value is larger than 1.0 and less than 1.0×10^9. For example, when the energy grid is 123.456789 eV, the notation of FRENDY and NJOY99 are 123.456789 eV and 1.234568+2 eV, respectively. Since the precision of the decimal notation is larger than that of the scientific notation when data value is larger than 1.0, FRENDY can treat the finer energy grid. For these reasons, the cross section of FRENDY is more precise than that of NJOY99 at the rapidly changing point. Since the difference is not so large, it does not affect the neutronics calculation in general.

< Figures 3-7 >

Table 1 shows the root mean square (RMS) value of the cross section relative

difference of ^{238}U and the average value of all nuclei prepared in JENDL-4.0 in all energy grids. Similar to the RMS value of ^{238}U, the RMS value of all nuclei is quite small. The main reason for the difference is attributable to the difference of number of energy grid. These results indicate that FRENDY appropriately processes the resonance reconstruction and Doppler broadening.

< Table 1 >

3.2. Verification of Processing Thermal Scattering Law Data

Comparisons of the coherent elastic and inelastic scattering cross sections for ^{1}H in liquid H_2O from JENDL-4.0 at 296 K are shown in **Figures 8 and 9**, respectively. The difference is only found at the discontinuous point in the coherent elastic cross section. **Figure 10** shows the detail of the discontinuity region of the coherent elastic scattering cross sections for ^{1}H in liquid H_2O. The differences are observed near the Blagg edge. The cause of the difference is the significant digits of the PENDF file and the rounding error of FRENDY and NJOY99.

< Figures 8-10 >

Table 2 shows the RMS value of the cross section relative difference of each material prepared in JENDL-4.0. The RMS value of the elastic scattering cross section is much larger than that of the inelastic scattering cross section. As shown in Figure 8, the elastic scattering cross section has large difference at the Bragg edge and this difference affects the RMS value.

The inelastic scattering cross section of underlined materials in Table 2, *i.e.*, para hydrogen, para deuterium and ortho deuterium, has large RMS value comparing with the other materials. **Figure 11** shows the comparisons of the inelastic scattering cross sections for ortho deuterium from JENDL-4.0 at 19 K. The large difference is observed in the low energy region and this difference affects the RMS value. The cause of the large difference in the low energy region was found to be a bug of NJOY99. **Figure 12** shows the inelastic scattering

cross section distribution of the ortho deuterium processed by NJOY99 when the incident energy is 1.78×10^{-5} eV. The symbol mark represents the secondary energy grid of NJOY99. Since the symbol mark is not observed from 1.5×10^{-6} eV to 1.7×10^{-5} eV, the cross section is not appropriately linearized at this energy region. This bug is also found in NJOY2012 [11]. Since this bug only affects the inelastic scattering cross section of limited materials and these materials are not generally used for the neutronics calculation, the difference of the inelastic scattering cross section has no large impact on the neutronics calculation except for special cases, *e.g.*, treatment of the cold neutron. It is difficult to find this bug if NJOY is only used for the nuclear data processing. Development of FRENDY is also useful for finding bugs and inappropriate processing in the other processing systems.

Other RMS values except the inelastic scattering cross section of above-mentioned materials are similar to H in liquid H_2O. These results indicate that FRENDY appropriately processes the thermal scattering law data.

< Table 2 and Figures 11-12 >

3.3. Comparison of Processing Time

To estimate the efficiency of FRENDY, the total processing time to generate the ACE file is compared with that of NJOY99. The GCC-4.4.7 [29] is used to compile FRENDY and NJOY99 and -O2 is used for the optimization.

Table 3 shows the total processing time to generate the ACE files in each nucleus and material prepared in JENDL-4.0 at 296 K. Note that tolerance value used for generating the ACE files is 0.1%. Though the calculation time of the probability table of FRENDY is much faster than that of NJOY99, the other calculation time of FRENDY is longer than that of NJOY99. Since the evaluated nuclear data file is not frequently processed, the processing time is not a serious problem for the neutronics calculation. Furthermore, from the view point of computer performance improvement, current processing time with FRENDY is acceptable

enough.

< Table 3 >

4. Comparison of k_{eff} values using ACE File Processed by FRENDY with those by NJOY99

To verify FRENDY, we compared the k_{eff} values of MCNP5 using ACE file of JENDL-4.0 prepared by FRENDY with those by NJOY99. The tolerance value to generate the ACE file is 0.1%.

To compare the k_{eff} values, 79 benchmark experiments, representing 752 critical configurations in the ICSBEP handbook [22] were used. The ICSBEP handbook contains hundreds of benchmark experiments, representing several thousand critical configurations. In this study, we selected benchmark experiments that are used in the validation of ENDF/B-VII.0 and ENDF/B-VII.1 [30, 31] for the verification of FRENDY. The list of benchmark experiments used in this verification, average values of a standard deviation and relative deference between FRENDY and NJOY99 are shown in **Tables 4 and 5**. We considered a lot of conditions and combinations of parameters as follows:

- Fissile materials (plutonium, highly/intermediate/low enriched uranium, ^{233}U, and mixed plutonium-uranium),
- Physical form of the fissile material (metal, compound and solution),
- Neutron energy range where the majority of the fissions occur (fast, intermediate, thermal, and mixed spectra systems).

MCNP sample input files in the ICSBEP handbook were used for the comparison of the k_{eff} values. Many of them were not intended to be used for the strict validation of the evaluated nuclear data file and neutronics calculation codes by the comparison of calculation results

with experimental results. In this verification, these sample input files are only used to compare the k_{eff} values using the ACE files processed by FRENDY with those by NJOY99.

< Tables 4-5 >

Tables 6 to 8 show the comparison of k_{eff} values for benchmark experiments using the ACE files processed by FRENDY and NJOY99. The benchmark name indicates fissile material, physical form of the fissile material, neutron energy range, and experimental number. For example, the meaning of "HCI005" is as follows:

H : fissile material is high enrichment Uranium,

C : physical form of the fissile material is compound,

I : neutron energy range where the majority of the fissions occur is intermediate

005 : experimental number is 5.

As shown in Tables 6 to 8, the relative difference does not depend on the fissile material, physical form of the fissile and neutron energy range and the differences are less than 4σ ($2\sigma_{FRENDY}+2\sigma_{NJOY99}$) in almost all critical configurations. Considering the standard deviation (1σ) of the k_{eff} value, the k_{eff} values using the ACE file processed by FRENDY is in good agreement with those by NJOY99. As described in the previous section, the difference of the cross section becomes large at the rapidly changing point. The comparison results indicate that the cross section difference does not affect the k_{eff} values. These results verify that FRENDY generates the ACE file correctly.

< Tables 6-8 >

5. Conclusions

We have developed a new nuclear data processing system FRENDY in order to process the evaluated nuclear data file immediately and independently when the new

evaluated nuclear data file is released. It is the world's first nuclear data processing system to generate the ACE file without NJOY modules. Though the current version of FRENDY supports the traditional ENDF-6 format only, it becomes possible to address the GND and other nuclear data formats if new parser, writer and converter classes are implemented. It has been developed by using modern programming techniques under careful consideration for maintainability, modularity, portability, and flexibility.

The results of FRENDY were compared with those of NJOY99 (updates 393). The comparison indicates that FRENDY appropriately processes resonance reconstruction, Doppler broadening, and thermal scattering law data. To compare the ACE files generated by FRENDY with those by NJOY99, 79 benchmark experiments, representing 752 critical configurations in the ICSBEP handbook were used. The k_{eff} values using the ACE file processed by FRENDY show good agreement with those by NJOY99. The relative difference does not depend on the type of the fissile material, physical form of the fissile and neutron energy range. These results verify that FRENDY generates the ACE file correctly.

The current version of FRENDY adopts similar data processing methods implemented in NJOY99. The improvement of processing methods and the new implementation of original functions will be investigated.

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Figure captions

Figure 1. The schematic diagram of FRENDY structure.

Figure 2. The development scheme of FRENDY.

Figure 3. The comparison of the total cross section for ^{238}U from JENDL-4.0 at 296 K.

Figure 4. The comparison of the elastic scattering cross section for ^{238}U from JENDL-4.0 at 296 K.

Figure 5. The comparison of the fission cross section for ^{238}U from JENDL-4.0 at 296 K.

Figure 6. The comparison of the radiative capture cross section for ^{238}U from JENDL-4.0 at 296 K.

Figure 7. The detail of the difference region of the fission cross section for ^{238}U from JENDL-4.0 at 296 K.

Figure 8. The comparison of the coherent elastic scattering cross section for ^{1}H in H_2O from JENDL-4.0 at 296 K.

Figure 9. The comparison of the inelastic scattering cross section for ^{1}H in H_2O from JENDL-4.0 at 296 K.

Figure 10. The detail of the discontinuity region of the inelastic scattering cross section for ^{1}H in H_2O from JENDL-4.0 at 296 K.

Figure 11. The comparison of the inelastic scattering cross section for ortho deuterium from JENDL-4.0 at 19 K.

Figure 12. The inelastic scattering cross section distribution when the incident energy is 1.78×10^{-5} eV for ortho deuterium from JENDL-4.0 processed by NJOY99.

Table 1. The RMS value of cross section relative difference of ^{238}U and average value of all nuclei prepared in JENDL-4.0 between FRENDY and NJOY99

	RMS* value				Energy grid no
	Total	Scatter	Fission	Radiative capture	
^{238}U	0.017%	0.017%	0.029%	0.033%	1,453,885
Average	0.071%	0.070%	0.007%	0.122%	109,607

*Root Mean Square

Table 2. The RMS value of cross section relative difference of each material prepared in JENDL-4.0 between FRENDY and NJOY99

	RMS* value		Temp. [K]	Energy grid no
	Elastic	Inelastic		
H in H_2O	3.133%	0.001%	296	2,599
Para hydrogen	3.196%	0.064%	20	2,599
Ortho hydrogen	3.196%	0.003%	20	2,599
H in ZrH	0.003%	0.010%	296	1,211
D in D_2O	3.071%	0.001%	296	2,734
Para deuterium	3.133%	0.099%	19	2,734
Ortho deuterium	3.133%	0.198%	19	2,734
Be	0.039%	0.001%	296	1,899
BeO	0.002%	0.001%	296	2,245
Graphite	0.002%	0.001%	296	2,219
Liquid methane	3.173%	0.001%	100	2,599
Solid methane	0.015%	0.006%	22	1,211
Polyethylene	0.003%	0.001%	296	1,211
Benzene	3.133%	0.001%	296	2,599
Zr in ZrH	0.003%	0.002%	296	7,119
Average	1.682%	0.026%		2,554
Average without under lined material	1.315%	0.002%		2,520

*Root Mean Square

Table 3. The total processing time and the ratio to generate the ACE file processed by FRENDY and NJOY99.

		Processing time [s]		FRENDY
		FRENDY	NJOY99	/ NJOY99
Nuclide (with probability table calculation)	^{1}H	0.1	0.2	0.5
	C	0.4	0.4	1.1
	^{56}Fe	24.8	8.5	2.9
	^{235}U	571.7	434.1	1.3
	^{238}U	439.1	429.4	1.0
	^{238}Pu	102.8	227.9	0.5
	^{239}Pu	269.5	279.6	1.0
	All	60657	85122	0.7
Nuclide (without probability table calculation)	^{1}H	0.1	0.2	0.5
	C	0.3	0.4	0.9
	^{56}Fe	21.0	8.6	2.4
	^{235}U	140.0	51.3	2.7
	^{238}U	327.0	195.0	1.7
	^{238}Pu	43.8	6.4	6.9
	^{239}Pu	128.0	53.0	2.4
	All	11440	2112	5.4
Thermal Scattering Law Data	H in H_2O	62.6	16.3	3.8
	H in ZrH	152.3	63.5	2.4
	Graphite	53.5	10.3	5.2
	Zr in ZrH	84.0	15.1	5.6
	All	1564	384	4.1

Table 4. The benchmark experiments used for the verification of FRENDY by using ICSBEP [21].

Fissile material	Chemical form of fuel	Average fission energy	Exp. No.	Critical configuration No.	Average of a standard deviation of MCNP5	Average of absolute value of relative difference
HEU (High enrichment Uranium)	METAL	FAST	1	2	0.014%	0.009%
			5	6	0.016%	0.027%
			7	43	0.016%	0.032%
			47	1	0.016%	0.048%
			49	3	0.014%	0.023%
			50	1	0.020%	0.009%
			72	2	0.014%	0.004%
			78	44	0.013%	0.020%
		INTER	6	3	0.017%	0.013%
		MIXED	1	2	0.019%	0.007%
		THERM (Thermal)	1	1	0.027%	0.109%
			6	23	0.018%	0.023%
	COMP	INTER	5	5	0.012%	0.007%
	SOL (Solution)	THERM	1	10	0.016%	0.018%
			2	14	0.016%	0.017%
			38	30	0.020%	0.022%
			42	8	0.010%	0.017%
			50	11	0.017%	0.017%
IEU (Intermediate enrichment Uranium)	COMP (Compound)	INTER (Intermediate)	1	4	0.014%	0.032%
		THERM	2	6	0.013%	0.020%
LEU (Low enrichment Uranium)	COMP	THERM	1	8	0.021%	0.036%
			2	5	0.024%	0.025%
			3	22	0.013%	0.018%
			6	18	0.013%	0.018%
			9	27	0.014%	0.016%
			10	30	0.014%	0.020%
			16	32	0.012%	0.016%
			17	3	0.012%	0.015%
			51	19	0.015%	0.023%
			79	10	0.014%	0.015%
	SOL	THERM	3	9	0.015%	0.027%
			4	7	0.013%	0.013%
			7	5	0.014%	0.027%
			16	7	0.016%	0.014%
			17	6	0.016%	0.020%
			18	6	0.015%	0.019%
			20	4	0.012%	0.024%
			21	4	0.013%	0.012%

Table 5. The benchmark experiments used for the verification of FRENDY by using ICSBEP [21] (cont.).

Fissile material	Chemical form of fuel	Average fission energy	Exp. No.	Critical configuration No.	Average of a standard deviation of MCNP5	Average of absolute value of relative difference
MIX	METAL	FAST	1	1	0.019%	0.017%
			2	3	0.015%	0.039%
	COMP	THERM	12	33	0.019%	0.036%
	SOL	THERM	1	12	0.016%	0.025%
			7	7	0.015%	0.022%
PU	METAL	FAST	1	4	0.007%	0.014%
			2	1	0.013%	0.021%
			5	1	0.014%	0.021%
			8	1	0.014%	0.001%
			9	1	0.014%	0.024%
			10	1	0.014%	0.016%
			11	1	0.017%	0.002%
			18	1	0.014%	0.003%
			44	5	0.014%	0.030%
			45	7	0.015%	0.010%
		INTER	2	1	0.012%	0.001%
	SOL	THERM	1	6	0.017%	0.031%
			2	7	0.016%	0.017%
			3	8	0.015%	0.019%
			4	13	0.014%	0.018%
			5	9	0.015%	0.020%
			6	3	0.014%	0.016%
			7	8	0.017%	0.027%
			8	29	0.016%	0.022%
			9	3	0.013%	0.038%
			10	14	0.016%	0.020%
			11	12	0.016%	0.018%
			12	22	0.012%	0.017%
			18	9	0.015%	0.012%
			21	10	0.016%	0.022%
			22	17	0.015%	0.014%
			34	15	0.015%	0.023%
U233	METAL	FAST	1	1	0.018%	0.057%
			2	2	0.019%	0.026%
			3	2	0.020%	0.005%
			4	2	0.019%	0.023%
			5	2	0.012%	0.015%
			6	1	0.012%	0.011%
	COMP	THERM	1	7	0.015%	0.014%
	SOL	THERM	12	8	0.013%	0.013%
			13	21	0.016%	0.021%

Table 6. Comparison of k_{eff} results for benchmark experiments using ACE files processed by FRENDY and NJOY99.

Bench-mark name	No.	FRENDY		NJOY99		Relative dif.	Relative dif. / $1\sigma_{NJOY99}$
		k_{eff}	Standard deviation ($1\sigma_{FRENDY}$)	k_{eff}	Standard deviation ($1\sigma_{NJOY99}$)	(FRENDY - NJOY99) / NJOY99	
HMF005	1	0.97241	0.014%	0.97267	0.014%	-0.027%	1.9
	2	0.98747	0.016%	0.98769	0.015%	-0.022%	1.5
	3	0.99805	0.016%	0.99790	0.015%	0.015%	1.0
	4	0.99555	0.016%	0.99530	0.016%	0.025%	1.6
	5	1.00784	0.016%	1.00781	0.015%	0.003%	0.2
	6	1.00339	0.016%	1.00332	0.015%	0.007%	0.5
HMI006	1	0.99358	0.017%	0.99370	0.018%	-0.012%	0.7
	2	0.99626	0.017%	0.99644	0.017%	-0.018%	1.1
	3	0.99861	0.017%	0.99869	0.018%	-0.008%	0.4
HMM001	1	1.07812	0.019%	1.07824	0.019%	-0.011%	0.6
	2	1.07738	0.019%	1.07742	0.019%	-0.004%	0.2
HMT006	1	0.99409	0.019%	0.99388	0.019%	0.021%	1.1
	2	0.99461	0.018%	0.99486	0.019%	-0.025%	1.3
	3	0.99908	0.019%	0.99918	0.018%	-0.010%	0.6
	4	0.99303	0.018%	0.99292	0.018%	0.011%	0.6
	5	0.99181	0.018%	0.99170	0.018%	0.011%	0.6
	6	0.98880	0.017%	0.98874	0.017%	0.006%	0.4
	7	0.98567	0.017%	0.98623	0.016%	-0.057%	3.5
	8	0.98090	0.016%	0.98088	0.016%	0.002%	0.1
HCI005	7	0.98970	0.012%	0.98971	0.012%	-0.001%	0.1
	9	1.17996	0.011%	1.17995	0.012%	0.001%	0.1
	10	1.19066	0.012%	1.19063	0.012%	0.003%	0.2
	15	1.43337	0.013%	1.43305	0.014%	0.022%	1.6
ICI001	18	0.97242	0.009%	0.97227	0.010%	0.015%	1.5
	19	0.98098	0.012%	0.98116	0.013%	-0.018%	1.4
	20	1.01781	0.016%	1.01846	0.017%	-0.064%	3.8
	21	0.94131	0.017%	0.94160	0.017%	-0.031%	1.8
ICT002	1	1.00369	0.013%	1.00351	0.014%	0.018%	1.3
	2	0.97880	0.014%	0.97848	0.015%	0.033%	2.2
	3	1.00952	0.014%	1.00935	0.013%	0.017%	1.3
	4	1.00407	0.013%	1.00397	0.013%	0.010%	0.8
	5	0.99349	0.013%	0.99331	0.013%	0.018%	1.4
	6	0.98684	0.013%	0.98659	0.014%	0.025%	1.8
LCT001	1	0.99965	0.020%	0.99949	0.021%	0.016%	0.8
	2	0.99725	0.022%	0.99811	0.021%	-0.086%	4.1
	3	0.99744	0.020%	0.99733	0.021%	0.011%	0.5
	4	0.99800	0.021%	0.99821	0.020%	-0.021%	1.1
	5	0.99589	0.020%	0.99573	0.021%	0.016%	0.8
	6	0.99792	0.021%	0.99794	0.021%	-0.002%	0.1
	7	0.99637	0.021%	0.99706	0.020%	-0.069%	3.5
	8	0.99636	0.021%	0.99568	0.020%	0.068%	3.4

Table 7. Comparison of k_{eff} results for benchmark experiments using ACE files processed by FRENDY and NJOY99 (cont.).

Bench-mark name	No.	FRENDY		NJOY99		Relative dif.	Relative dif. / $1\sigma_{NJOY99}$
		k_{eff}	Standard deviation ($1\sigma_{FRENDY}$)	k_{eff}	Standard deviation ($1\sigma_{NJOY99}$)	(FRENDY - NJOY99) / NJOY99	
MCT012	1	0.91526	0.018%	0.91545	0.019%	-0.021%	1.1
	2	0.91589	0.018%	0.91588	0.018%	0.001%	0.1
	3	0.91458	0.018%	0.91447	0.018%	0.012%	0.7
	4	0.91596	0.018%	0.91658	0.018%	-0.068%	3.8
	5	0.91508	0.018%	0.91510	0.019%	-0.002%	0.1
	6	0.91780	0.019%	0.91807	0.018%	-0.029%	1.6
	7	0.98987	0.017%	0.98989	0.018%	-0.002%	0.1
	8	0.98656	0.017%	0.98687	0.018%	-0.031%	1.7
MMF001	1	0.99715	0.019%	0.99732	0.019%	-0.017%	0.9
PMF001	1	0.99904	0.007%	0.99887	0.007%	0.017%	2.4
	2	0.99942	0.007%	0.99965	0.007%	-0.023%	3.3
	3	0.99927	0.006%	0.99918	0.006%	0.009%	1.5
	4	1.00018	0.006%	1.00011	0.006%	0.007%	1.2
PMI002	1	1.00453	0.012%	1.00454	0.012%	-0.001%	0.1
UCT001	1	1.00315	0.015%	1.00329	0.015%	-0.014%	0.9
	2	0.99969	0.016%	0.99983	0.017%	-0.014%	0.8
	3	0.99574	0.016%	0.99573	0.016%	0.001%	0.1
	4	1.00206	0.013%	1.00231	0.013%	-0.025%	1.9
	5	1.00589	0.013%	1.00581	0.013%	0.008%	0.6
	7	1.00106	0.016%	1.00121	0.016%	-0.015%	0.9
	8	1.00162	0.014%	1.00144	0.014%	0.018%	1.3
UMF001	1	0.99944	0.018%	0.99887	0.019%	0.057%	3.0
UMF005	1	0.99633	0.011%	0.99610	0.011%	0.023%	2.1
	2	0.99585	0.012%	0.99591	0.012%	-0.006%	0.5

Table 8. Comparison of k_{eff} results for benchmark experiments using ACE files processed by FRENDY and NJOY99 (cont.).

Bench-mark name	No.	FRENDY		NJOY99		Relative dif.	Relative dif. / $1\sigma_{NJOY99}$
		k_{eff}	Standard deviation ($1\sigma_{FRENDY}$)	k_{eff}	Standard deviation ($1\sigma_{NJOY99}$)	(FRENDY - NJOY99) / NJOY99	
HST001	1	1.00086	0.017%	1.00114	0.017%	-0.028%	1.6
	2	0.99788	0.017%	0.99794	0.017%	-0.006%	0.4
	3	1.00495	0.016%	1.00478	0.017%	0.017%	1.0
	4	1.00107	0.017%	1.00074	0.017%	0.033%	1.9
	5	1.00271	0.015%	1.00292	0.014%	-0.021%	1.5
	6	1.00658	0.015%	1.00639	0.015%	0.019%	1.3
	7	1.00091	0.017%	1.00093	0.017%	-0.002%	0.1
	8	1.00115	0.017%	1.00117	0.017%	-0.002%	0.1
	9	0.99645	0.017%	0.99679	0.017%	-0.034%	2.0
	10	0.99656	0.014%	0.99638	0.015%	0.018%	1.2
LST004	1	0.99968	0.015%	0.99969	0.015%	-0.001%	0.1
	29	1.00058	0.014%	1.00032	0.014%	0.026%	1.9
	33	0.99809	0.014%	0.99802	0.014%	0.007%	0.5
	34	1.00009	0.013%	1.00039	0.013%	-0.030%	2.3
	46	1.00005	0.013%	0.99999	0.013%	0.006%	0.5
	51	0.99898	0.012%	0.99907	0.012%	-0.009%	0.8
	54	0.99913	0.012%	0.99901	0.012%	0.012%	1.0
PST001	1	1.00666	0.016%	1.00646	0.016%	0.020%	1.2
	2	1.00716	0.017%	1.00780	0.016%	-0.064%	4.0
	3	1.00944	0.017%	1.00961	0.017%	-0.017%	1.0
	4	1.00387	0.017%	1.00354	0.017%	0.033%	1.9
	5	1.00776	0.017%	1.00740	0.017%	0.036%	2.1
	6	1.00819	0.017%	1.00800	0.017%	0.019%	1.1
UST012	1	0.99496	0.014%	0.99514	0.014%	-0.018%	1.3
	2	0.99454	0.014%	0.99449	0.014%	0.005%	0.4
	3	1.00400	0.014%	1.00392	0.014%	0.008%	0.6
	4	0.99798	0.014%	0.99815	0.013%	-0.017%	1.3
	5	1.00063	0.013%	1.00053	0.013%	0.010%	0.8
	6	1.00209	0.013%	1.00221	0.013%	-0.012%	0.9
	7	1.00013	0.011%	0.99994	0.011%	0.019%	1.7
	8	0.99722	0.012%	0.99710	0.012%	0.012%	1.0

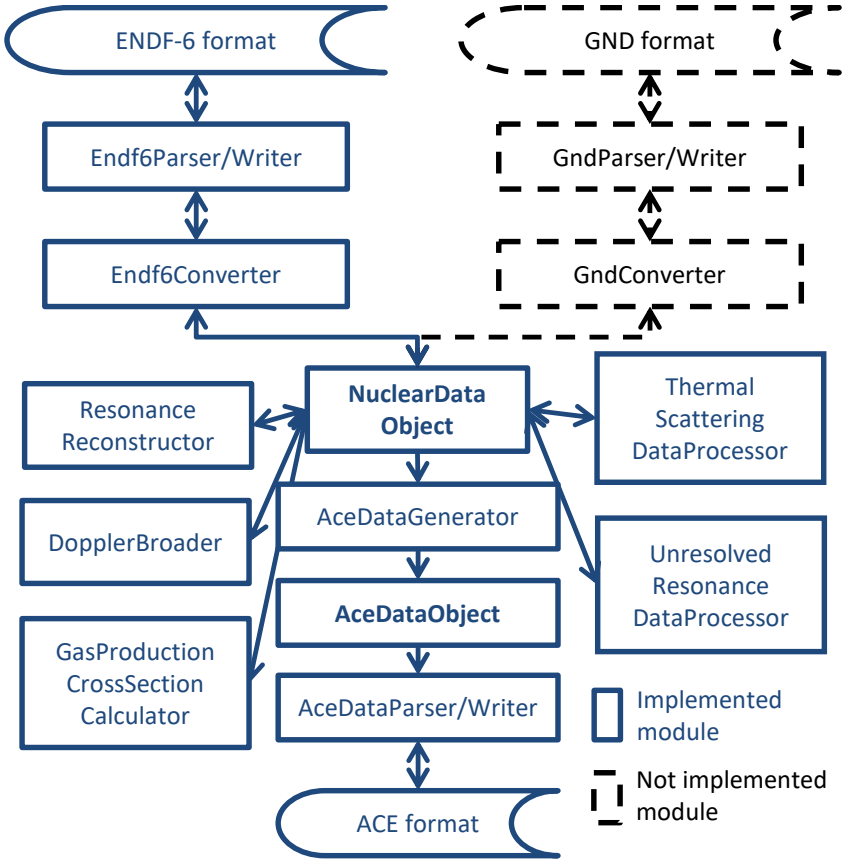

Figure 1. The schematic diagram of FRENDY structure.

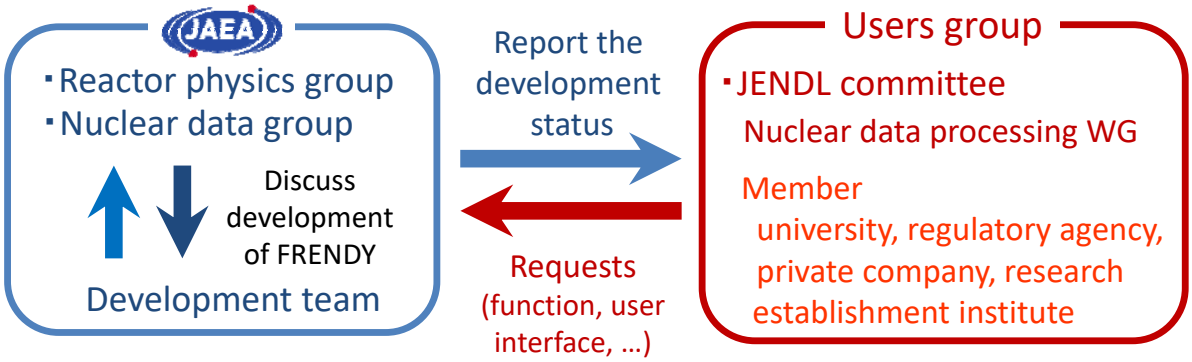

Figure 2. The development scheme of FRENDY.

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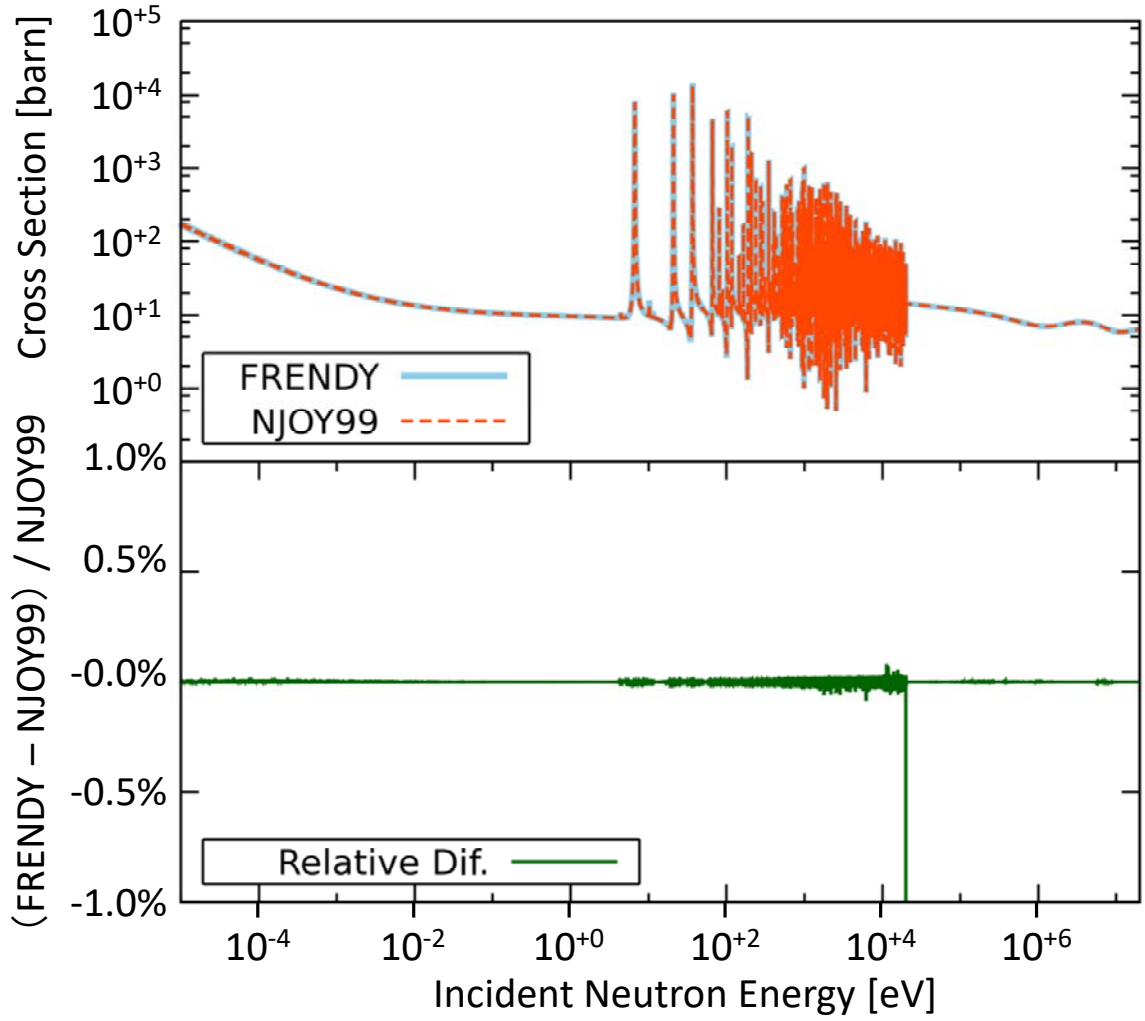

Figure 3. The comparison of the total cross section for ^{238}U from JENDL-4.0 at 296 K.

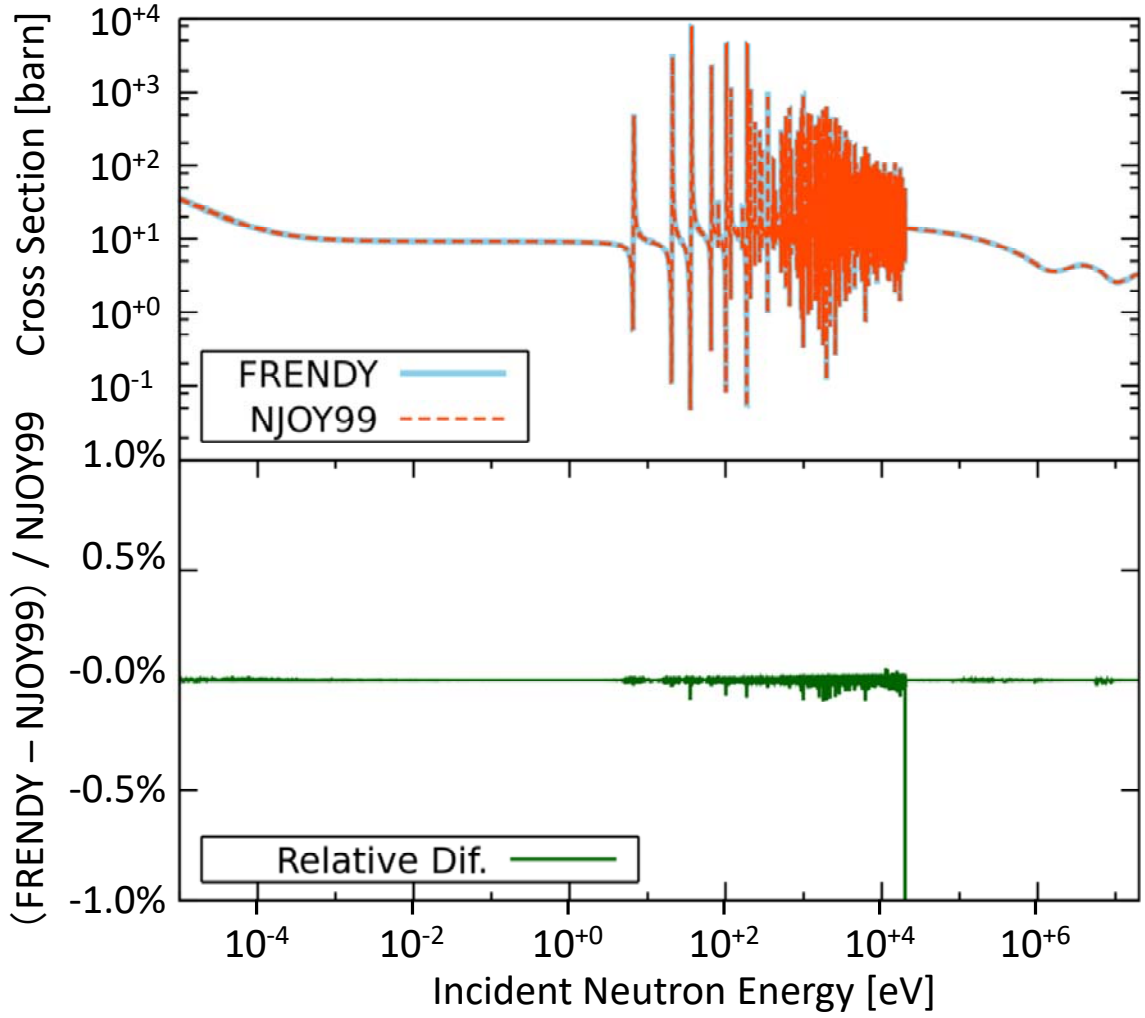

Figure 4. The comparison of the elastic scattering cross section for ^{238}U from JENDL-4.0 at 296 K.

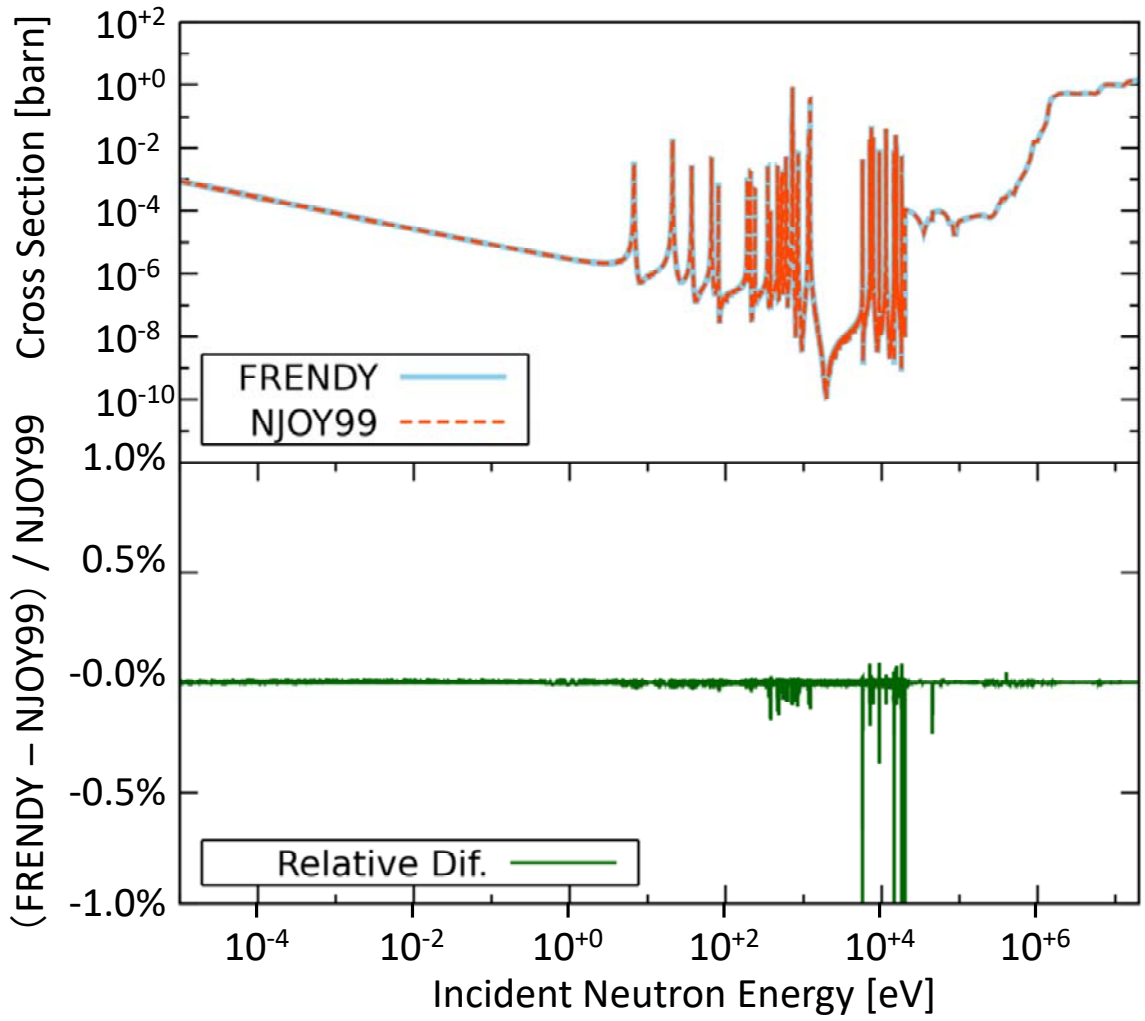

Figure 5. The comparison of the fission cross section for ^{238}U from JENDL-4.0 at 296 K.

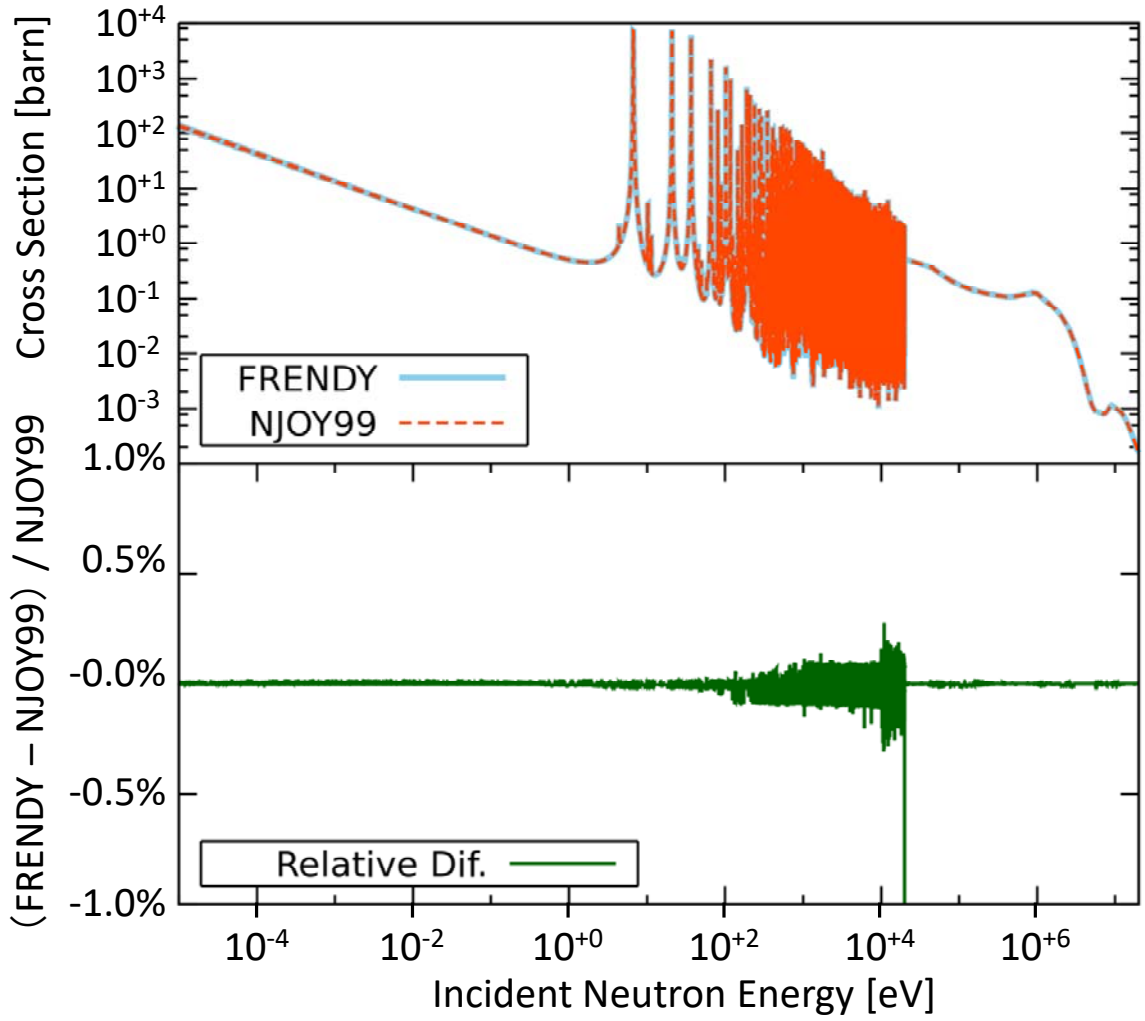

Figure 6. The comparison of the radiative capture cross section for ^{238}U from JENDL-4.0 at 296 K.

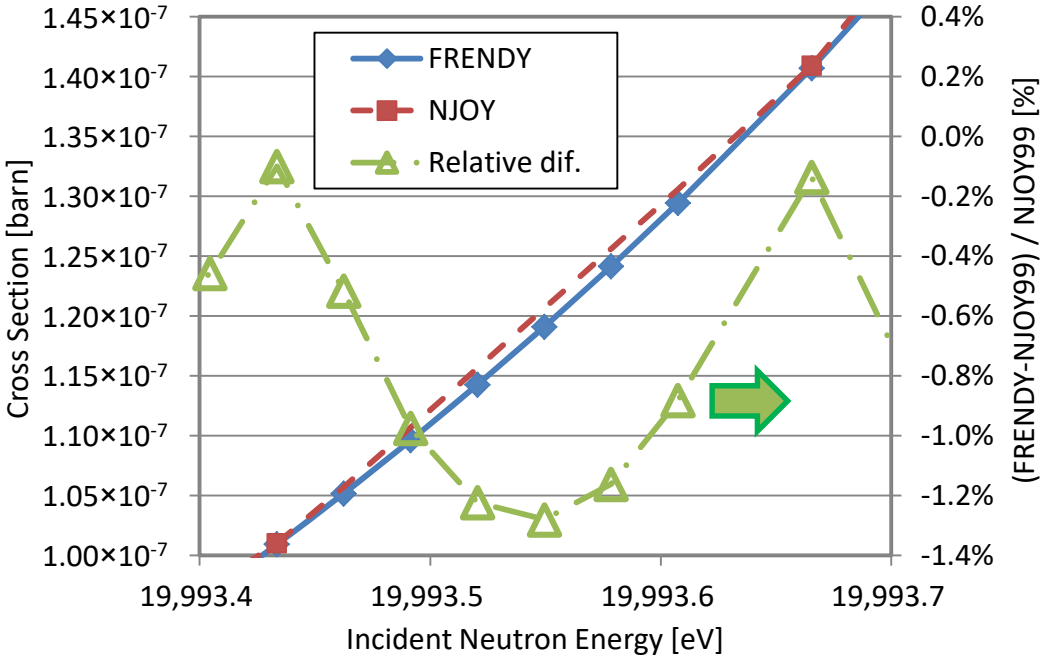

Figure 7. The detail of the difference region of the fission cross section for ^{238}U from JENDL-4.0 at 296 K.

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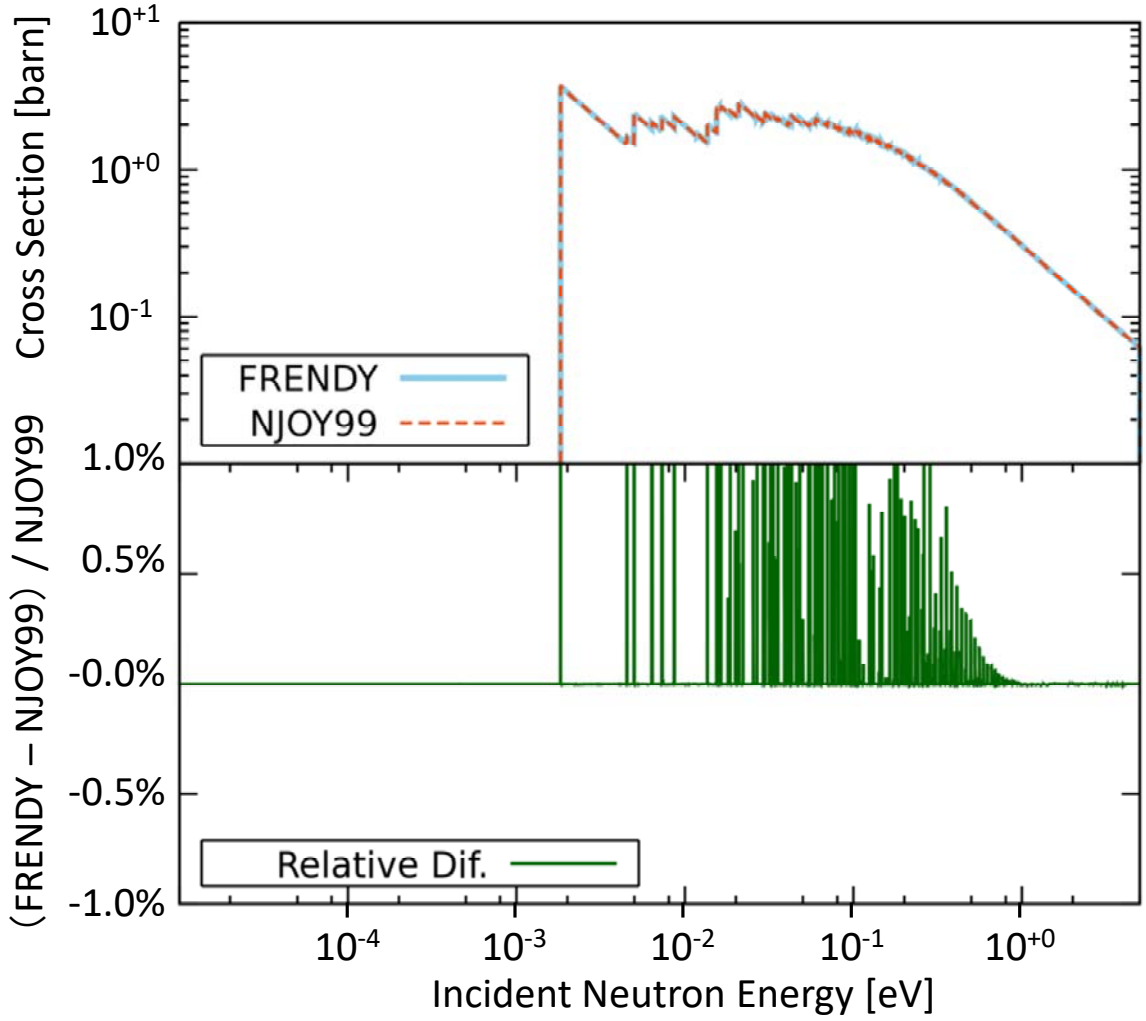

Figure 8. The comparison of the coherent elastic scattering cross section for ^{1}H in H_2O from JENDL-4.0 at 296 K.

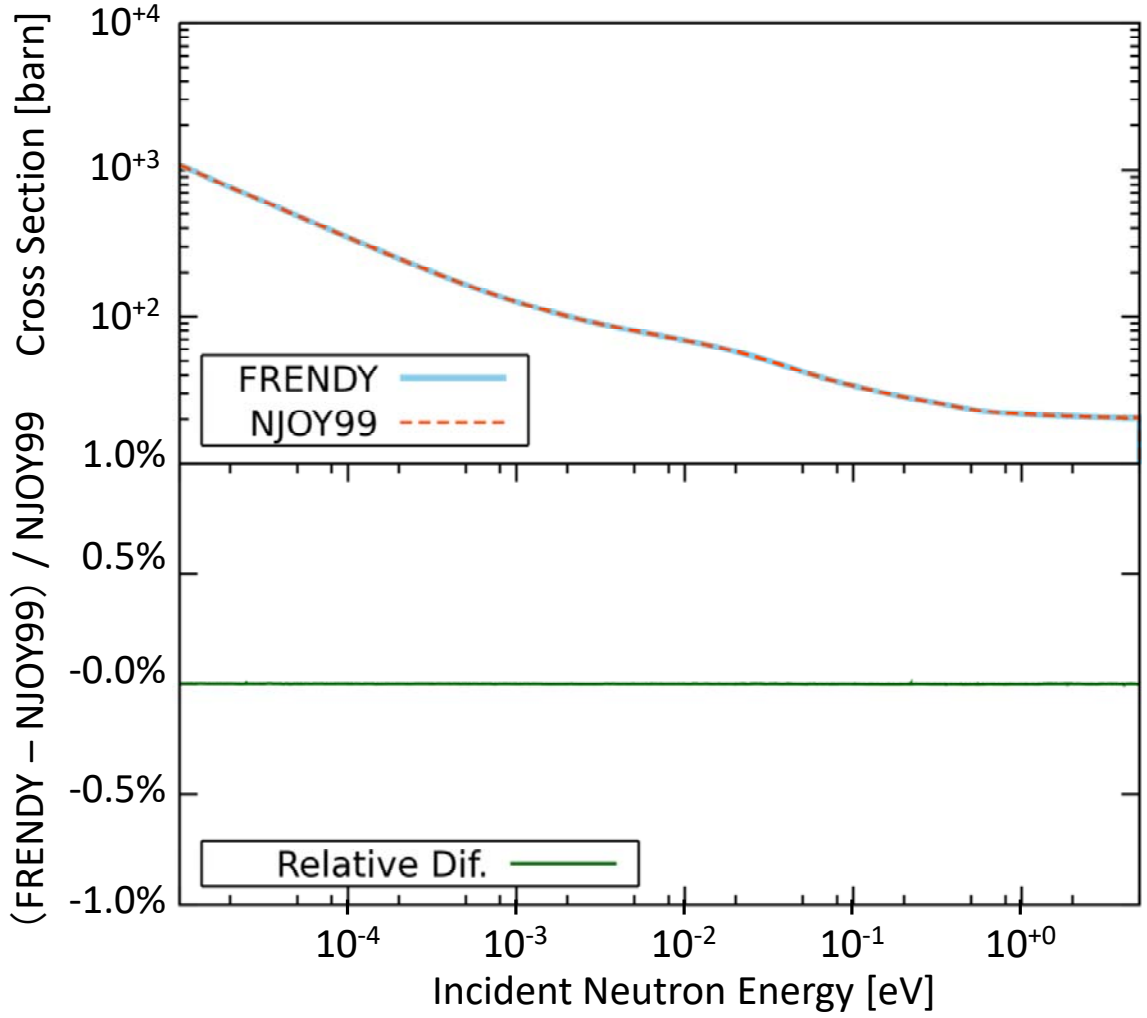

Figure 9. The comparison of the inelastic scattering cross section for ^{1}H in H_2O from JENDL-4.0 at 296 K.

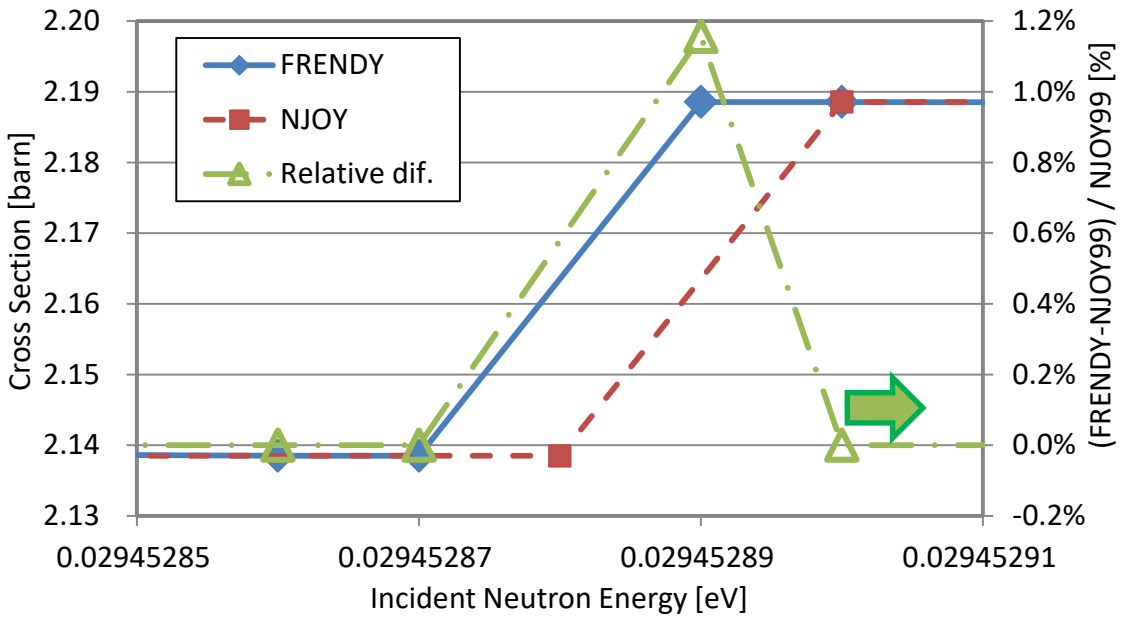

Figure 10. The detail of the discontinuity region of the coherent elastic scattering cross section for ^{1}H in H_2O from JENDL-4.0 at 296 K.

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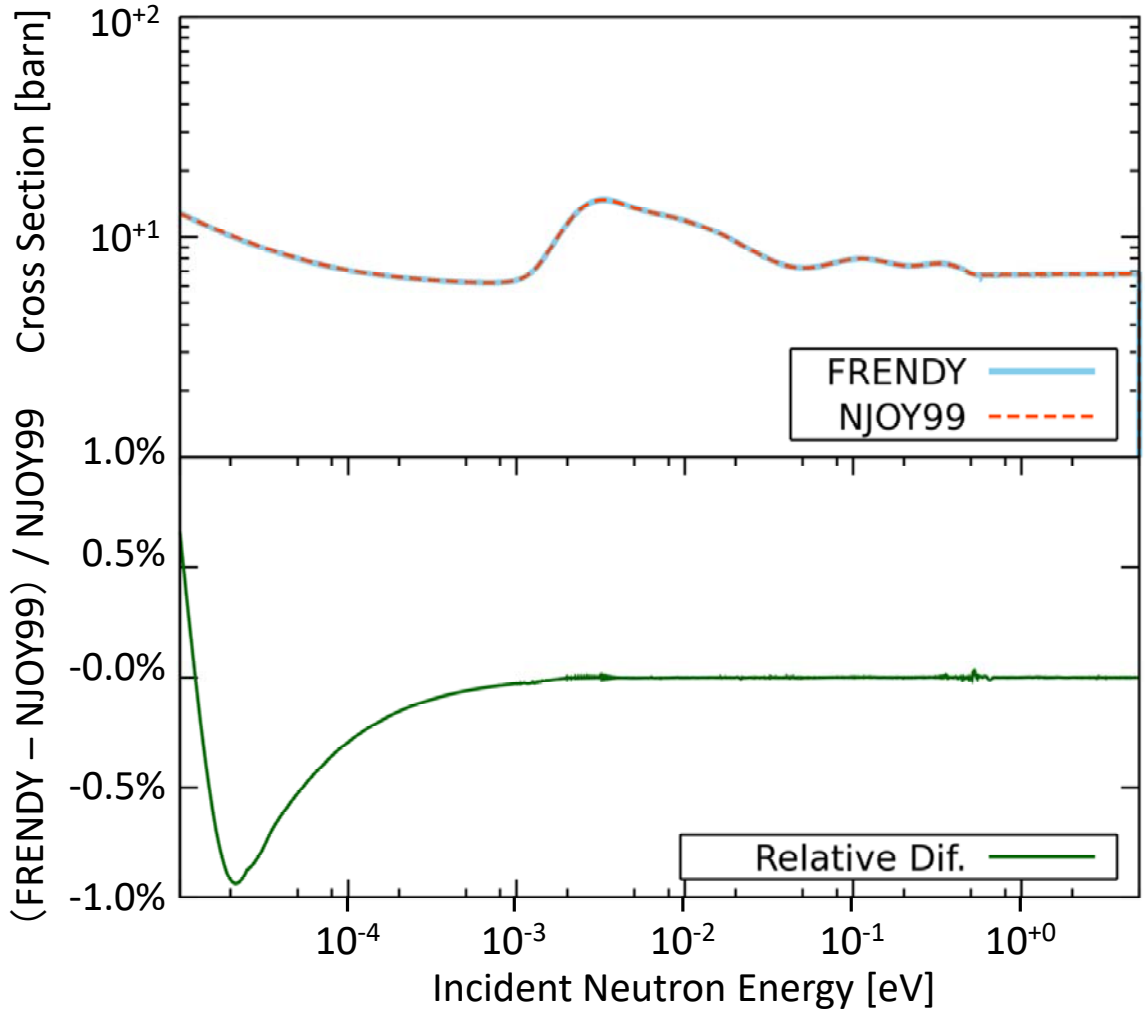

Figure 11. The comparison of the inelastic scattering cross section for ortho deuterium from JENDL-4.0 at 19 K.

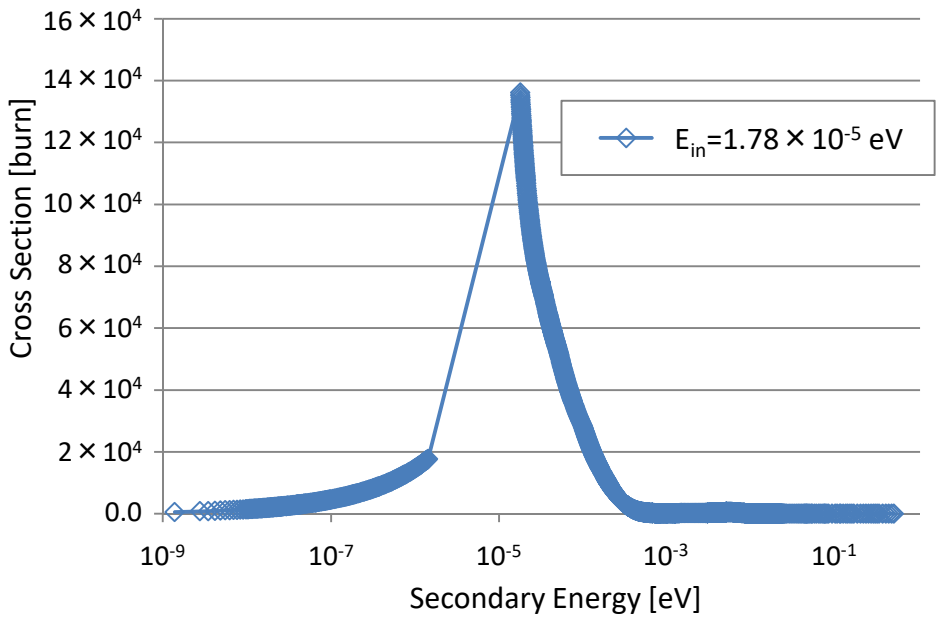

Figure 12. The inelastic scattering cross section distribution when the incident energy is 1.78×10^{-5} eV for ortho deuterium from JENDL-4.0 processed by NJOY99.