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Status of the JENDL General-Purpose File

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Abstract

It is about 30 years since we released the first version of JENDL General-Purpose File, which was developed in cooperation with the Japanese Nuclear Data Committee. We have continued to update the libraries in order to meet users' requirements. The latest version JENDL-3.3 is being used in various application fields, and its reliability has been confirmed.

We are now working on the new library JENDL-4 as a mid-term plan for the Japan Atomic Energy Agency. Nuclear model codes have been developed to raise the reliability of JENDL-4. Resolved resonance parameters were re-examined and updated by considering new measurements. In the FP mass region, the coupled-channel optical model parameters were searched for in order to perform data evaluation in the non-resonance energy range. As for MA, fission and capture cross sections were evaluated. Covariance estimation has just started in the MA mass region, although we produced covariances of several nuclides in JENDL-3.3 for applications to a design study on ADS.

KEYWORDS: Nuclear Data, Evaluation, Cross Section, Covariance, JENDL

1. Introduction

Versions of Japanese Evaluated Nuclear Data Library (JENDL) have been developed in cooperation with the Japanese Nuclear Data Committee (JNDC) since early 1970s. We still continue to update the library in order to improve its reliability and satisfy users' requirements. The latest version, JENDL-3.3 [1], which was released in 2002, has been used in various application fields. After the release of JENDL-3.3, we surveyed nuclear data needs in the *Ad Hoc* Committee on Next JENDL, which was founded in JNDC. The committee concluded that the new general-purpose file JENDL-4 should be developed for R&D on innovative reactors, the high burn-up and use of MOX fuels for LWR, criticality safety, and basic sciences. At the integration of two research organizations, former Japan Atomic Energy Research Institute (JAERI) and Japan Nuclear Cycle Development Institute (JNC), in October 2005, the Nuclear Data Center, Japan Atomic Energy Agency (JAEA), officially announced a mid-term plan to create JENDL-4. In the new library, much emphasis is placed on improvements of fission product (FP) and minor actinide (MA) data which are becoming more important than before. JENDL-4 is scheduled to be released

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in FY2009 after integral validations.

This paper deals with the developments of JENDL. To begin with, the history of JENDL is described in Chapter 2. Secondly, the integral performance of the latest version JENDL-3.3 is given in Chapter 3. Activities for the new library JENDL-4 are described in Chapter 4.

2. History of JENDL

The characteristics of JENDL General-Purpose Files are given in Table 1. The driving force to initiate the JENDL evaluations was the fast reactor project in Japan. People involved in the project felt that the national library would be requisite to analyze the integral experiments for the MONJU reactor. The first version JENDL-1 [2] was released in 1977 for FBR. The second version JENDL-2 [3] was completed in 1982 for applications to LWR and FBR. JENDL-2 had drawbacks when it was applied to fusion neutronics, although it had a good predicting power in other fields. That is the reason why we started a series of JENDL-3 evaluations. Gamma-ray production data were evaluated to calculate nuclear heating for JENDL-3.1. The latest version JENDL-3.3 contains covariances for 20 nuclides and double-differential neutron emission spectra for 60 nuclides. The covariance data are much more required than before in order to estimate the design accuracy of nuclear facilities from the viewpoints of safety and economics.

Table 1: Characteristics of JENDL.

	JENDL-1[2]	JENDL-2[3]	JENDL-3.1[4]	JENDL-3.2[5]	JENDL-3.3[1]
Year	1977	1982	1990	1994	2002
Application	FBR	FBR, LWR	General	General	General
Max. energy	15 MeV	20 MeV	20 MeV	20 MeV	20 MeV
No. of nuclides(a)	66+6	173+8	305+19	318+22	335+2
γ production(b)	0	0	59	66	114
Covariances(c)	0	0	1	1	20
DDX(d)	0	0	0	0	60

(a) No. of isotopes + no. of natural elements
(b) No. of nuclides containing secondary γ-ray production
(c) No. of nuclides containing covariances
(d) No. of nuclides containing double-differential neutron spectra

3. Performances of JENDL-3.3

In JENDL-3.3, the major problems of the previous library were resolved. The overestimate of k_{eff} for U-fueled thermal-neutron cores, which was the biggest problem with JENDL-3.2, was removed by changing the resolved resonance parameters and prompt fission neutron spectra for ^{235}U. Figure 1 shows the C/E values of k_{eff} for thermal cores. It is found from the figure that the JENDL-3.3 calculations are in good agreement with the measurements at medium enriched U fuel cores of STACY, TRACY and JRR-4. For the MOX fuel cores, there is no big difference between the JENDL-3.2 and JENDL-3.3 values, both of which are in better agreement with the measurements than the ENDF/B-VI calculations.

As for the fast-neutron cores shown in Fig. 2, we cannot see a big difference between the JENDL-3.3 and -3.2 calculations except for the ^{233}U fuel cores, *i.e.*, JEZEBEL-23 and FLATTOP-23. The fission cross section of ^{233}U was reduced in the fast-neutron energy region as a result of the simultaneous evaluation [6]. This fact led to a significant improvement of the C/E values for JEZEBEL-23 and FLATTOP-23 with the JENDL-3.3 data.

Figure 1: C/E of k_{eff} for thermal-neutron cores.

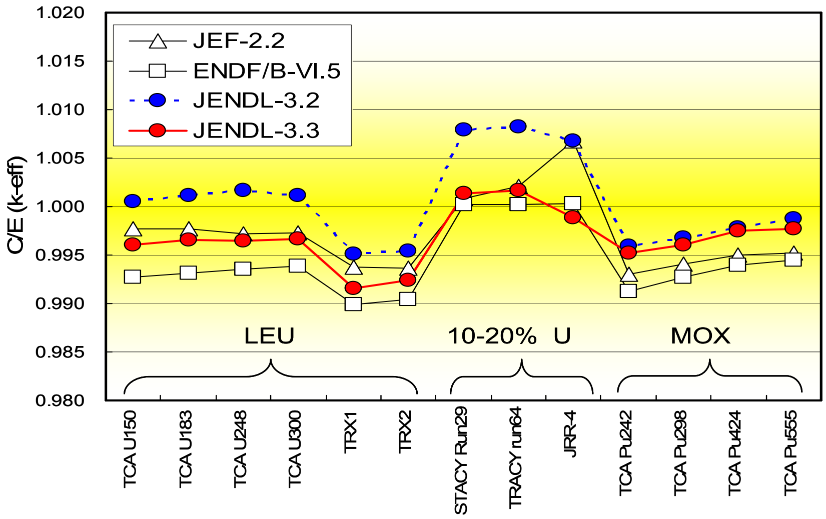
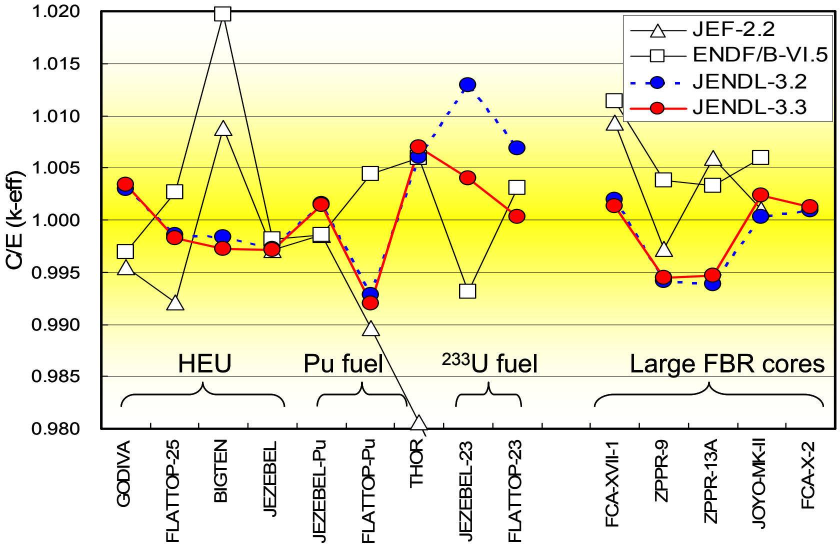


Figure 2: C/E of k_{eff} for fast-neutron cores.

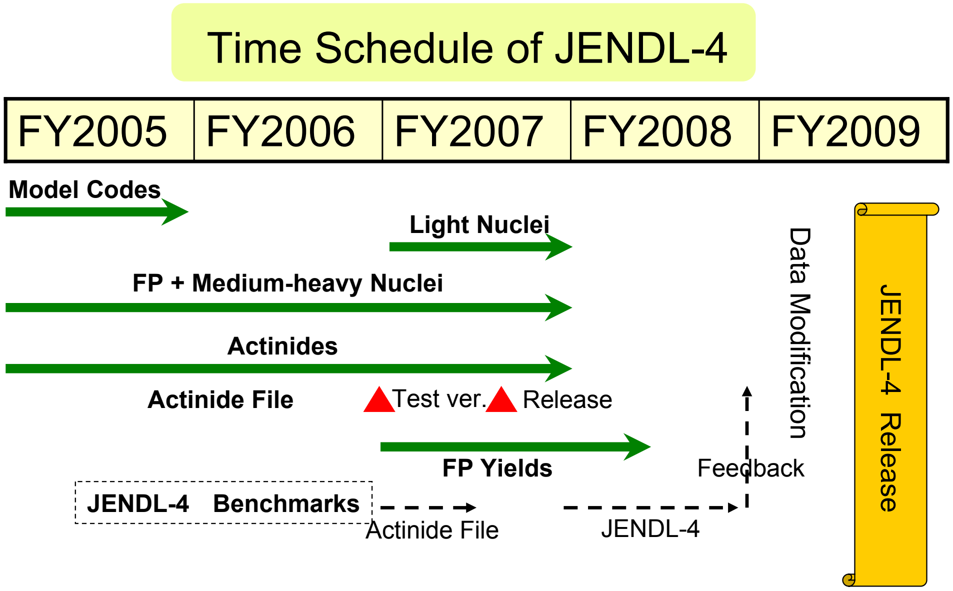


4. Activities for JENDL-4

4.1 Time Schedule

Figure 3 shows the time schedule of JENDL-4. We plan to release the JENDL Actinide File which is a special purpose file in FY2007, because advanced users need to utilize new data before the release of JENDL-4. A year will be spent in the benchmark tests of the new library.

Figure 3: Time Schedule of JENDL-4.



4.2 Key Subjects

We still have open problems of JENDL-3.3. Criticalities are slightly underestimated for thermal cores with low-enriched U fuel. As for fast reactors, it was pointed out [7] that somewhat large capture cross sections might lead to the underestimation of criticalities for BFS cores. These problems should be resolved.

MA data become more important than before, since the development of ADS and the high burn-up and use of MOX fuels need more accurate MA data. For the nuclear fuel-cycle applications such as transport and storage of spent fuels, more accurate FP data are required to consider the criticality safety with the use of burn-up credit. The data for long-lived FP (LLFP) are also crucial for transmutation using ADS or FBR.

Covariances are needed for many applications. JENDL-3.3 contains covariance data for 20 nuclei. However, the number is insufficient. Covariances for MA and FP data should be estimated from measurements or nuclear model calculations.

Re-examination of FP yield data has been requested by many users. In JENDL-3.3, most of the yield data were taken from the JNDC FP Decay Data File Version-2 [8]. Fission neutron spectra including spontaneous fission should be evaluated for MA.

It is necessary to evaluate gamma-ray production data for more nuclei. Gamma-ray production with MOX fuels is more significant than that with UO_2 fuels. Model calculations should be carried out for evaluation of gamma-ray production data for FP nuclides where few measurements are available.

For a radiation damage study, emitted charged-particle and PKA spectra are required to calculate KERMA factors and DPA cross sections. For light nuclei in JENDL-3.3, there exist few charged-particle spectra except for ^{2}H and ^{9}Be. The helium production due to ^{59}Ni becomes significant for PWR cores ten years after irradiation. Therefore, the data should be evaluated for JENDL-4.

It is required to raise the reliability of JENDL-4 better than that of JENDL-3.3. We have to make criteria of data validation from the viewpoints of quality assurance. Benchmark tests are performed for various applications. The results of the benchmark tests are fed back to evaluations. We should provide users with the databases composed of input and output of benchmark calculations together with data on integral experiments, which enable users to repeat calculations. This is important to disseminate JENDL over nuclear industries. Furthermore, for users' convenience, reactor constants must be produced for many applications. It is believed that these activities help to make JENDL-4 a standard nuclear data library in Japan.

4.3 Development of Nuclear Model Codes

As mentioned above, evaluations of MA and FP data are important for JENDL-4. However, experimental data are scarce for these nuclides. Thus, nuclear model calculations play a significant role in evaluation. We have resumed making nuclear model codes in order to reflect recent advances in nuclear theories on evaluations. This would make future extensions of codes easier.

Independent developments are being performed for medium-heavy and actinide nuclides, since a systematics is searched for in each mass region.

Figure 5: ^{93}Nb(n,n') at 25.7 MeV.

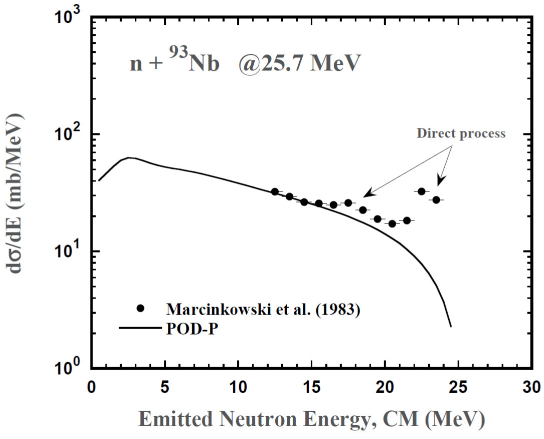
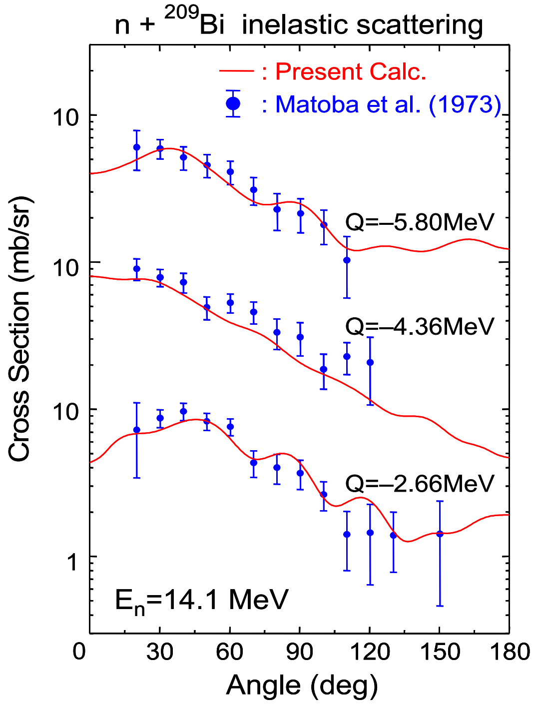


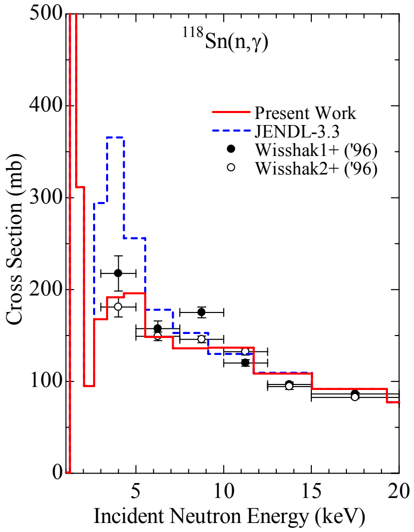
Figure 4: ^{209}Bi(n,n') at 14.1 MeV.



The codes are based on the spherical and coupled-channel optical models, the distorted-wave Born approximation (DWBA), the pre-equilibrium exciton model, and the Hauser-Feshbach statistical model with width fluctuation corrections.

Figure 4 shows an example of DWBA calculations with the POD code [9], angular distributions of neutrons inelastically scattered from ^{209}Bi. Pre-equilibrium neutron emission spectra are illustrated in Fig. 5 for the n+^{93}Nb reaction at 25.7 MeV.

Figure 6: ^{118}Sn(n,γ) cross section.



4.4 FP Data

4.4.1 Resolved Resonance Parameters

Resolved resonance parameters of JENDL-3.3 FP nuclides were examined for JENDL-4 by taking account of recent measurements. As a result, the parameters for 89 FP nuclides were updated. Moreover, the parameters for additional 13 nuclides were newly evaluated for JENDL-4. As for ^{99}Tc, we obtained a thermal value of 23.6±0.7 b for the capture cross section from the latest measurements [10-12], and revised the parameters for a negative resonance. Figure 6 shows the energy-averaged capture cross section of ^{118}Sn, of which resonance parameters were taken from the data measured by Wisshak et al. [13] In the case of ^{118}Sn, the upper limit of the resolved resonance was extended to 15 keV from 4.8 keV in JENDL-3.3.

4.4.2 Optical Model Parameters

In the non-resonance energy region, nuclear model calculations are performed to evaluate cross sections. Although many parameters are required as input to nuclear model codes, the most important ones are optical model parameters. The optical model parameters were searched for by using the coupled-channel optical

Figure 7: Total cross sections.

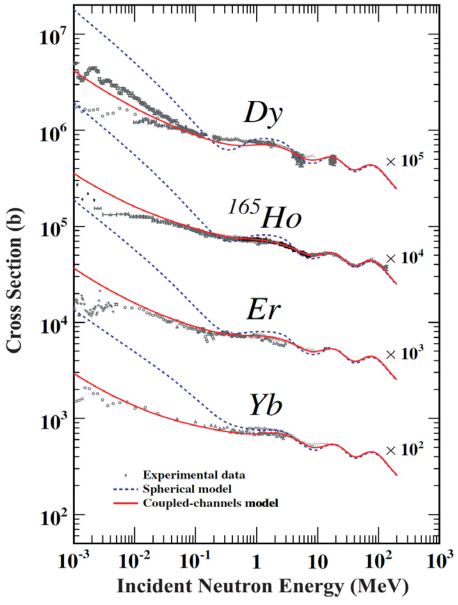
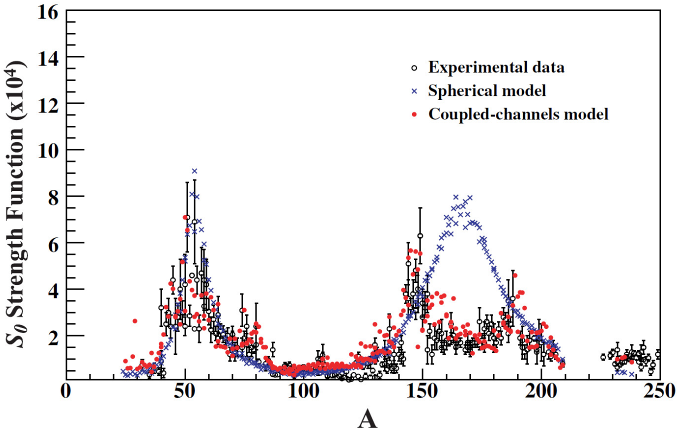


Figure 8: s-wave neutron strength functions.



model code OPTMAN [14]. The systematics of neutron and proton optical model parameters has been obtained by comparing experimental data on Fe, Ni, Cu, Zr, Nb, Mo, Sn, W, Au, Pb, Bi, Th, and U in the energy region from 1 keV to 200 MeV. The calculated total cross sections are illustrated in Fig. 7, where the spherical optical model calculations are also shown. It is found from the figure that the present calculations reproduce experimental data very well. In order to check the low-energy behavior of the optical model parameters, we also calculated neutron strength functions at 10 keV, which represent average resonance properties. As seen in Fig. 8, the present calculations are in good agreement with experimental data. The spherical optical model cannot reproduce the experimental data in the mass region from A=150 to 200.

4.5 MA Data

4.5.1 Resolved Resonance Parameters

Resolved resonance parameters have been updated for ^{236}Pu, ^{242,242m}Am, and ^{242,243,244,245,246,247,248}Cm, and ^{250}Cf. For example, the fission cross section of ^{243}Cm is shown in Fig. 9. The presently evaluated data reproduce the experimental data of Silbert et al. [15] The calculated thermal fission cross section, 587.4 b, agrees with the weighted average of experimental data, 587±12 b.

4.5.2 Fission and Capture Cross Sections

We have performed the evaluation of fission cross sections on the basis of available experimental data. The least-squares fitting code GMA [16] was used for this purpose. The GMA analyses were performed for 23 MA nuclei: ^{230,232}Th, ^{231}Pa, ^{232,234,236}U, ^{237}Np, ^{236,238,242,244}Pu, ^{241,242m,243}Am, ^{242,243,244,245,246,247,248}Cm, and ^{249,252}Cf. The present results are almost the same as the evaluated data in JENDL-3.3, and significant discrepancies were not observed.

The capture cross section of ^{237}Np has been evaluated by using the statistical model code CCSM [17]. The calculated cross sections are in good agreement with available experimental data, as seen in Fig. 10.

4.6 Medium-Heavy Nuclide Data

As for medium-heavy nuclei, we have evaluated stable Ca isotopes using the statistical model code TNG [18]. Most of the parameters required were taken from

Figure 9: ^{243}Cm(n,f) cross section.

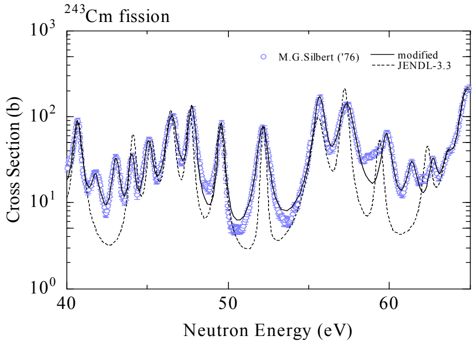


Figure 10: ^{237}Np(n,γ) cross section.

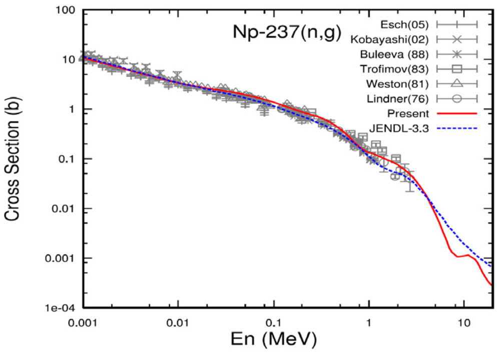
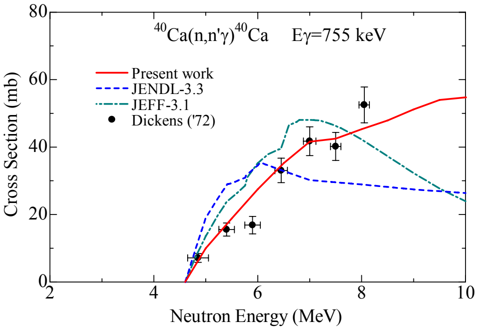


Figure 11: ^{40}Ca(n,n'γ)^{40}Ca cross section.



the Reference Input Parameter Library RIPL-2 [19] which was developed under the auspices of IAEA. The level density parameters are based on the work of Mengoni and Nakajima [20]. It is found from Fig. 11 that the calculated gamma-ray production cross sections agree with the ^{40}Ca(n,nγ)^{40}Ca data measured by Dickens [21].

4.7 Covariances

Covariances can be estimated from experimental data or nuclear model calculations on which the evaluated data are based. The least-squares fitting code GMA [16] is capable of producing covariance data together with mean values. In the case of nuclear model calculations, cross-section covariances are calculated by the propagation of the covariances of model parameters, such as optical potentials and level density.

At present, no covariance data are fixed for JENDL-4. However, we made covariances of ^{15}N, ^{206,207,208}Pb, ^{209}Bi, ^{237}Np, ^{238}Pu, $^{241, 242m, 243}$Am and ^{244}Cm in JENDL-3.3 for a design study on ADS. The capture cross section of ^{243}Am is illustrated in Fig. 12 together with standard deviations, which were obtained from the covariance evaluation system KALMAN developed by Kawano and Shibata [22]. The covariances of the elastic scattering cross section of ^{15}N were estimated from the uncertainties in the optical model parameters and the level density parameter. The relative standard deviations are shown in Fig. 13.

Figure 12: ^{243}Am(n,γ) cross section.

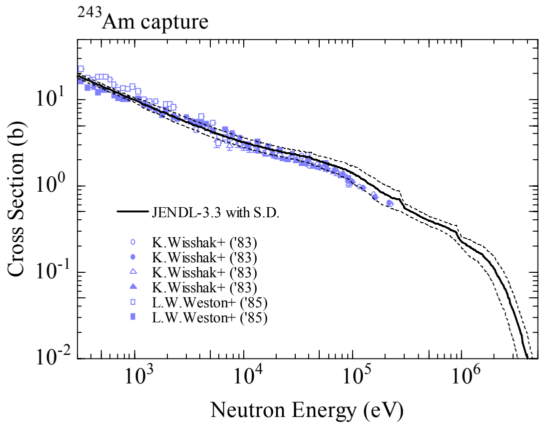
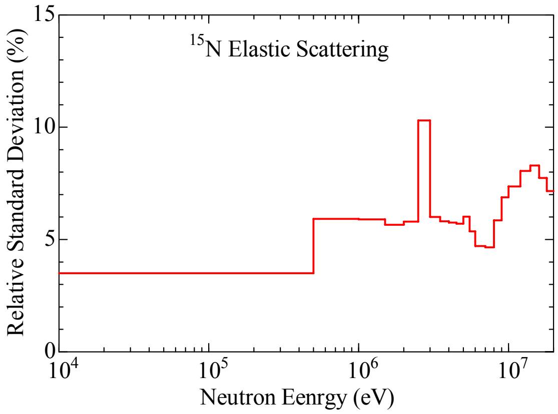


Figure 13: ^{15}N(n,n) cross section.



5. Conclusions

It is about 30 years since the development of JENDL was started by the requirements from the Japan's fast reactor project. We still continue to update the JENDL General-Purpose File in order to raise its reliability and satisfy users' demands. The latest version JENDL-3.3 has been applied to many fields and its reliability was confirmed. After the integration of JAERI and JNC, the Nuclear Data Center issued a mid-term plan in which the new library JENDL-4 is developed until the end of FY2009. In the new library, much emphasis is placed on MA and FP data. Developments of nuclear model codes have been performed to raise the reliability of the new library. Evaluation work is in progress for JENDL-4. We have already obtained more reliable evaluated data than JENDL-3.3 especially in the FP and MA mass region.

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