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EVALUATIONS OF HEAVY NUCLIDE DATA FOR JENDL-3.3

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New evaluations of neutron nuclear data for Uranium, Plutonium, and Thorium isotopes which are essential for applications to nuclear technology were carried out for the Japanese Evaluated Nuclear Data Library, JENDL-3.3. The objectives of the current release of JENDL were to fix several problems which have been reported for the previous version, to improve the accuracy of the data, and to evaluate covariances for the important nuclides. Quantities in JENDL-3.2 were extensively re-evaluated or replaced by more reliable values. The heavy nuclide data in JENDL-3.3 were validated with several benchmark tests, and it was reported that the current release gave a good prediction of criticalities.

Keywords: JENDL-3.3, Cross Section, Neutron, Evaluation, Uranium, Plutonium, Thorium, Resonance Parameter, Simultaneous Evaluation, Prompt Fission Neutron Spectrum, Direct/Semidirect Capture, Number of Prompt Neutrons, Number of Delayed Neutrons, Covariance

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JENDL-3.3 のための重核データの評価

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原子力技術開発において重要なウラン、プルトニウム、トリウムの同位体に対する中性子核データの新たな評価を行った。この評価値は日本の評価核データライブラリである JENDL-3.3 の一部となる。この評価の主たる目的は、前ヴァージョンに対して報告されていた幾つかの問題点の解決、データの精度向上、主要核種に対する共分散の評価、である。JENDL-3.2 に格納されている種々の核データを検討し、その多くについて再評価を行うか、もしくはより信頼できる数値に置き直した。JENDL-3.3 の重核データに対して種々のベンチマークテストが行われ、臨界性予測精度は以前の JENDL よりも向上していることが報告された。

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1 INTRODUCTION

Japanese Evaluated Nuclear Data Library (JENDL) version 3[1] was released in 1989, and JENDL-3.2[2] appeared in 1994. Since then a number of problems concerning the version 3.2 have been reported by users of the data library. One of the most serious problem for the heavy nuclide data was an overestimation of criticalities of thermal reactors when JENDL-3.2 was used.

In 1996, those problems were investigated by a special committee which was set up in Japanese Nuclear Data Committee (JNDC), and a new working group — Heavy Nuclide Data Evaluation Working Group — was organized to update the evaluated data of Uranium, Plutonium, and Thorium isotopes in 1998. The objectives of the current release of JENDL (JENDL-3.3[3]) were to fix those problems, to improve the accuracy of the data, and to evaluate covariances for the important nuclides.

Data in JENDL-3.2 were extensively re-evaluated or replaced by more reliable values. Those were resonance parameters for ^{235}U, ^{240}Pu, and ^{232}Th, secondary neutron energy spectra calculated with the GNASH code[4], prompt fission neutron spectra obtained by the multimodal analyses[5, 6] as well as an inclusion of pre-equilibrium effect[7], fission cross sections obtained by a new simultaneous evaluation[8], capture cross sections calculated with the DSD model[9, 10], and so on. The new library was released in May 2002 as JENDL-3.3. Talbes 1–5 summarize the changes of nuclear data of ^{232}Th, ^{233}U,^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu.

The covariance matrices were given for the cross sections, ν_d, ν_p, resonance parameters, Legendre-polynomial coefficients for elastic scattering, and prompt fission neutron spectra of Uranium and Plutonium isotopes. The covariances of the resolved resonance parameters were approximated by means of a simplified manner[11], if the covariance given by the R-matrix analysis was not available.

Several benchmark tests have been performed with the new library[3], and they reported that JENDL-3.3 improves the prediction of criticalities. In this report we present how the evaluation of heavy nuclide data was performed under the Heavy Nuclide Data Evaluation Working Group. Methods of evaluation, a theoretical background of model calculations, computer codes developed in our project, and parameters used here are reviewed, and comarisons with the previous evaluation are shown. A brief report was already published in Ref. [12].

Table 1: Summary of revision in JENDL-3.3, MF=1; number of neutrons per fission. The sign "yes" represents that the value was updated in JENDL-3.3, and the blank field means that the value in JENDL-3.2 was adopted.

MT		^{232}Th	^{233}U	^{235}U	^{236}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
452	$\overline{\nu}$		yes	yes		yes				
455	$\overline{\nu}_d$		yes	yes		yes				
456	$\overline{\nu}_p$		yes	yes						

Table 2: Summary of revision in JENDL-3.3, MF=2; resonance parameters.

LRU		^{232}Th	^{233}U	^{235}U	^{236}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
1	Resolved	yes		yes				yes		yes
2	Unresolved		yes	yes			yes	yes		

Table 3: Summary of revision in JENDL-3.3, MF=3; cross sections.

MT		^{232}Th	^{233}U	^{235}U	^{236}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
1	Total			yes						
2	Elastic	yes	yes	yes	yes	yes	yes	yes	yes	yes
16	$2n$		yes	yes		yes		yes		yes
17	$3n$		yes	yes		yes		yes		yes
18	Fission	yes	yes	yes		yes	yes	yes	yes	
37	$4n$	n.g.	n.g.	yes	n.g.	yes		yes		yes
51–90	Inelastic					yes		yes		yes
91	Continuum Inelastic					yes		yes		yes
102	Capture	yes	yes	yes	yes	yes	yes	yes	yes	yes

n.g. : not given

Table 4: Summary of revision in JENDL-3.3, MF=4; angular distribution.

MT		^{232}Th	^{233}U	^{235}U	^{236}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
2	Elastic	yes				yes				
16	$2n$									
17	$3n$									
18	Fission									
37	$4n$	n.g.	n.g.		n.g.	yes				
51–90	Inelastic							yes		yes
91	Continuum Inelastic					yes				

n.g. : not given

Table 5: Summary of revision in JENDL-3.3, MF=5; energy distribution.

MT		^{232}Th	^{233}U	^{235}U	^{236}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
16	$2n$	yes	yes	yes	yes	yes	yes	yes	yes	yes
17	$3n$	yes	yes	yes	yes	yes	yes	yes	yes	yes
18	Fission			yes		yes	yes			
37	$4n$	n.g.	n.g.		n.g.	yes				
91	Continuum Inelastic	yes	yes	yes	yes	yes	yes	yes	yes	yes
455	$\overline{\nu}_d$	yes	yes	yes	yes	yes	yes	yes	yes	yes

n.g. : not given

2 Data Evaluation Procedure

2.1 Resolved Resonance Parameters

2.1.1 ^{235}U

In JENDL-3.1, resonance parameters of the single-level Breit-Wigner (SLBW) formula was adopted for ^{235}U in the energy range up to 100 eV[13]. These parameters were superseded by the data of ENDF/B-VI which were based on a Reich-Moore (RM) R-matrix analysis of Leal, de Saussure, and Perez[14] at the release of JENDL-3.2. Note that Leal *et al.* gave the resonance parameters up to 2.25 keV, however the upper limit of the resolved resonance region for ^{235}U in JENDL-3.2 was 500 eV. Several problems concerning cross sections in the resolved resonance region have been reported since then. The major problems are an overestimation of k_{eff} for thermal reactors with highly enriched fuel. Of course there are several quantities which account for the problem, but we surveyed the resonance parameters first since they are strongly related to the thermal fission properties, and decided to employ the recent resonance parameter set obtained by the ORNL group[15]. Thermal cross sections and resonance integral calculated with those parameters are shown in Table 6.

A benchmark test with the resonance parameters of Leal *et al.*[15] was carried out, and it was reported that the prediction of k_{eff} was improved to some extent, but it was still insufficient. To improve the predictability of k_{eff}, re-evaluation of other quantities such as a prompt neutron fission spectrum at thermal energy was needed.

2.1.2 ^{232}Th

Another improvement of nuclear data in the resonance region is the parameters for ^{232}Th. The resonance parameters at the negative energy and scattering radius R' were adjusted and $1/v$-background cross section to the capture cross section was revised so as to reproduce the following data simultaneously:

- Capture cross section of 7.40 b at the thermal energy.
- Experimental capture cross sections of Chrien *et al.*[16], and Little *et al.*[17]
- Experimental total cross sections of Little *et al.*[17] and Kobayashi *et al.*[18]

The scattering radius R' in JENDL-3.2 is 10.25 fm, and a newly assigned value in JENDL-3.3 is 10.01 fm. Figure 1 shows the comparison of total and capture cross sections calculated with the resonance parameters in JENDL-3.2 and JENDL-3.3. The difference in the total cross section between JENDL-3.2 and JENDL-3.3 is small. However, the capture cross sections in JENDL-3.3 are larger than those of JENDL-3.2 near 10 eV. Since we adopted the thermal capture cross section of 7.40 b, the JENDL-3.3 capture cross section below 4 eV is larger than the experimental data of Little *et al.*[17]. A Th-core integral test at Kyoto University Critical Assembly (KUCA) reported that the capture cross section in JENDL-3.2 was underestimated. The increase in the capture cross section is consistent with the result of this integral test, and with the experimental cross sections.

2.1.3 ^{240}Pu

For ^{240}Pu a new Reich-Moore resonance parameter set was obtained by Bouland *et al.*[19] This R-matrix parameter set was adopted in JENDL-3.3, however the capture widths of -3 and 1.056 eV resonances were slightly modified in order to improve an integral test for some MOX fuel cores[20] and to reproduce the capture cross section at 0.0253 eV recommended by Mughabghab[21]. The following capture widths were adopted:

E_R [eV]	Γ_γ [eV]	
	Bouland *et al.*	JENDL-3.3
-3	0.039098	0.026
1.056	0.029148	0.030

With this modification the capture cross sections near 1 eV increase slightly, as shown in Fig. 2. The neutron capture cross section at the thermal energy becomes almost the same as that in JENDL-3.2, which is 289 b. In Fig. 3 the cross sections are expressed as the ratio to those in JENDL-3.2.

2.1.4 ^{242}Pu

Resolved resonances from 1.696 keV to 1.891 keV were newly added. These resonance energies and neutron widths were taken from the compilation of Mughabghab[21]. The capture widths of those resonances were determined by a systematic study of Murata[22], and the fission widths were from fission area data[23, 24]. Some other resonance parameters in JENDL-3.2 were revised in the same way.

Table 6: Fission and capture cross sections at the thermal energy calculated with the ^{235}U resonance parameters.

		Fission		Capture		Ref.
	formula	2200 m/s [b]	Res. Integ. [b]	2200 m/s [b]	Res. Integ. [b]	
JENDL-3.1	SLBW	584.0	275	96.0	152	[13]
JENDL-3.2	R-M	584.4	279	98.8	134	[14]
JENDL-3.3	R-M	585.1	276	98.7	141	

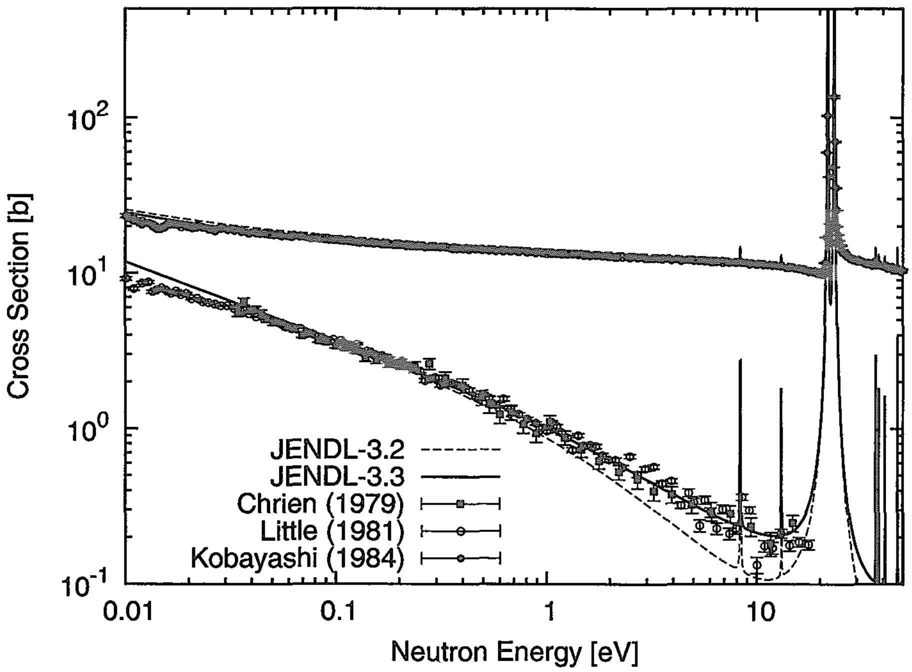

Fig. 1: Comparison of the total and capture cross sections of ^{232}Th with the experimental data and the evaluated cross sections in JENDL-3.2 and JENDL-3.3.

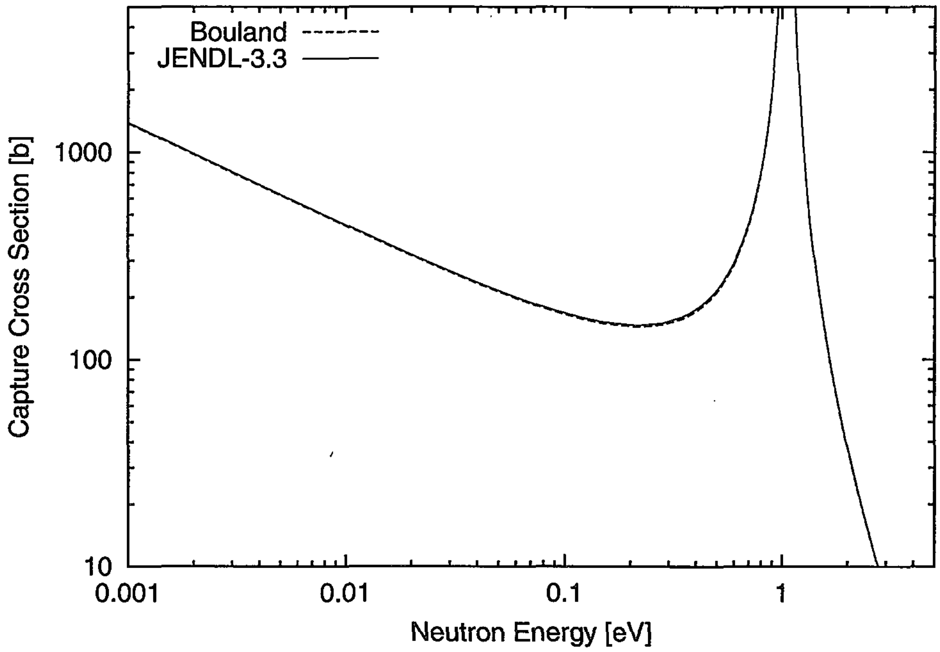

Fig. 2: Capture cross section of ^{240}Pu in the low energy region. The dashed line is the point-wise cross section calculated with the resonance parameters of Bouland *et al.*[19], and the solid line is with the modified resonance parameters.

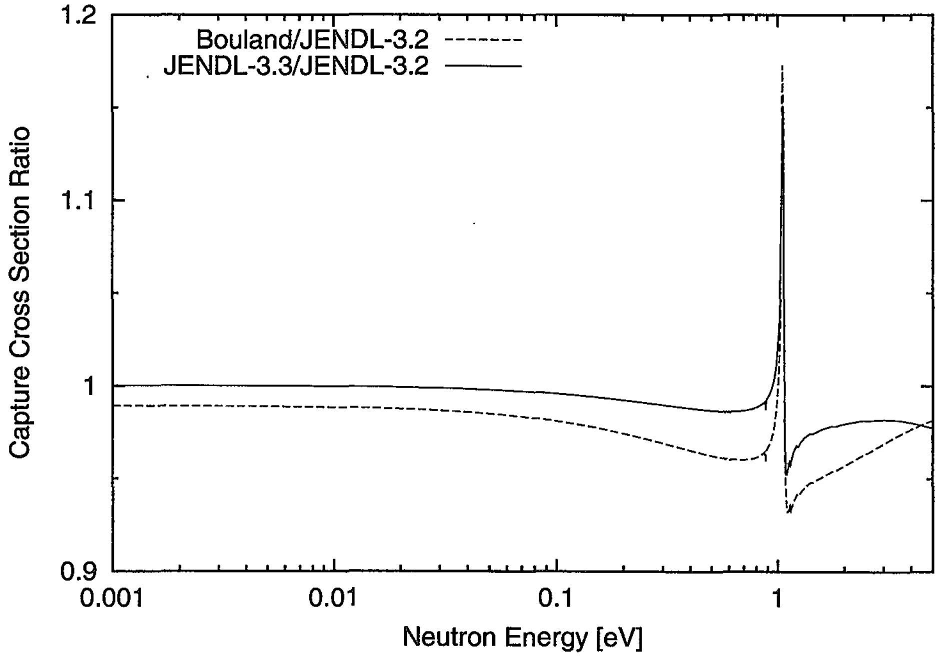

Fig. 3: Ratio of the ^{240}Pu capture cross sections to the JENDL-3.2 values.

2.2 Unresolved Resonance Parameters

The unresolved resonance parameters of ^{233}U were given in the energy region from 150 eV to 30 keV. The total cross section in this energy range was determined from the experimental data of Fulwood *et al.*[25] and Stupegia[26], and the fission cross section above 5 keV from Gwin *et al.*[27]. The capture cross section above 10 keV was calculated from the measured α-values of Hopkins and Diven[28] and the fission cross section of Gwin *et al.*[27]. The other cross sections, the capture below 10 keV and the fission below 5 keV, were taken from JENDL-3.2. The unresolved resonance parameters were determined so as to reproduce those cross sections.

The upper boundary of the resolved resonance region of ^{235}U was chosen as 2.25 keV, and the unresolved resonance parameters were given above this energy up to 30 keV. The total cross section was determined from the experimental data of Uttley *et al.*[29]. The fission cross section was based on the data of Weston and Todd[30]. The capture cross section was obtained from the α-values measured by Corvi *et al.*[31] and the fission cross section of Weston and Todd. The unresolved resonance parameters were determined from those cross sections.

For ^{239}Pu, the parameters at 30 keV were slightly modified to smoothly connect the cross sections.

The unresolved resonance region of ^{240}Pu is given in the energy region from 2.7 to 40 keV. The energy dependent unresolved resonance parameters were determined to reproduce evaluated cross sections based on the data of Weston and Todd[32]. Although background cross sections were given in JENDL-3.2, the revised parameters reproduce all cross sections without the background cross sections.

2.3 Simultaneous Evaluation of Fission Cross Sections

A new simultaneous evaluation[8] of fission cross sections for Uranium and Plutonium isotopes was carried out for JENDL-3.3. The simultaneous evaluation[33, 34] was already adopted to evaluate the fission cross sections of ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, and ^{241}Pu for JENDL-3[1]. However, some new measurements of the fission cross sections of these nuclides have been published since then. Those new information should be included in the modern evaluation in order to make our evaluation more reliable. In addition, some modifications were made for the result of the original simultaneous evaluation when they were adopted to JENDL-3.2[2], therefore a consistency among the evaluated data was lost, and its covariance data were no longer appropriate. For example, the fission cross section of ^{235}U above 14 MeV was modified independently for JENDL-3.2. Those facts became a driving-force to carry out the new simultaneous evaluation for JENDL-3.3.

The experimental data used were taken from the database EXFOR. The absolute and relative measurements of ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, and ^{241}Pu fission cross sections were carefully selected and compiled into a database for the present evaluation. In the evaluation of JENDL-3[1], capture cross sections of ^{238}U and ^{197}Au were incorporated into the database. These reactions were, however, omitted in the present evaluation, because the evaluation of the ^{238}U capture cross section was independent[35] of the simultaneous evaluation, and there is no data for ^{197}Au in JENDL. Our experimental database with the complete references is reported elsewhere[36]. To perform the simultaneous evaluation we developed the computer program SOK[36], which is a FORTRAN77 code based on the program KALMAN[37].

A remarkable feature in the new evaluation is that we included the ^{233}U data which were not used in the previous simultaneous evaluation[33, 34]. A comparison of the evaluated fission cross sections of ^{233}U with the evaluated values in JENDL-3.2 and ENDF/B-VI, as well as the experimental data is shown in Fig. 4. Figure 5 shows a comparison of the fission cross section ratios of ^{233}U to ^{235}U. The experimental data which have been considered are all shown by the same symbol (○). The evaluated ^{233}U fission cross sections are smaller than those in JENDL-3.2, and this is favorable for integral tests because an overestimation of k_{eff} has been reported for fast neutron spectrum cores with ^{233}U. In fact a new benchmark test[38] reported that the new ^{233}U data improved a prediction of criticalities for the ^{233}U cores of JEZEBEL-23 and FLATTOP-23.

In our evaluation, the fission cross sections for ^{235}U were determined in the energy range 30 keV – 20 MeV. A similar simultaneous evaluation[39] was performed for ENDF/B-VI, and it was reported that the uncertainties of the cross sections were very small. Although those two evaluations insist that their uncertainties are small, a systematic difference can be seen in the energy range 1–4 MeV, and the difference is about

2–3% which is larger than the uncertainties accompanying those evaluations. A ratio of the fission cross sections in JENDL-3.3 to ENDF/B-VI is shown in Fig. 6. In this figure the cross sections are represented by a 640-group structure.

The difference of 2–3% is not so large if one sees how experimental data are scattered in that energy region. However ^{235}U fission cross sections near 2 MeV are very sensitive to reactor calculations. This problem was investigated[40] in detail as a part of an international cooperation for evaluations of standard cross sections. Spectrum averaged cross sections

$$\overline{\sigma}_f = \int_0^\infty \chi(E)\sigma_f(E)dE, \tag{1}$$

where $\chi(E)$ is the well-defined neutron spectrum, were calculated with the evaluated fission cross sections, and comparisons with the measured values were made. The spectra used were, the ^{235}U prompt fission neutron spectrum at thermal energy, the ^{252}Cf spontaneous fission neutron spectrum, and the neutron spectrum produced by the ^{9}Be(d, xn) reaction. For the ^{235}U prompt fission neutron spectrum, the ENDF/B-VI evaluation reproduces the experimental data. For the ^{252}Cf and ^{9}Be(d, xn) neutron spectra, the JENDL-3.3 evaluation gives better results than ENDF/B-VI. The C/E values for those integral data are summarized in Table 7, which was taken from Ref. [40]. Results of a criticality benchmark test[38] for GODIVA are also shown in Table 7.

Table 7: C/E values for the spectrum averaged fission cross sections[38, 40]. The results of GODIVA criticality benchmark were calculated with the continuous energy Monte Carlo code MVP.

Neutron Field	JENDL-3.3	ENDF/B-VI
^{235}U prompt fission	1.015	0.997
^{252}Cf spontaneous fission	1.004	0.987
^{9}Be(d, xn) reaction	1.007	0.988
GODIVA criticality benchmark	1.0032	0.9965

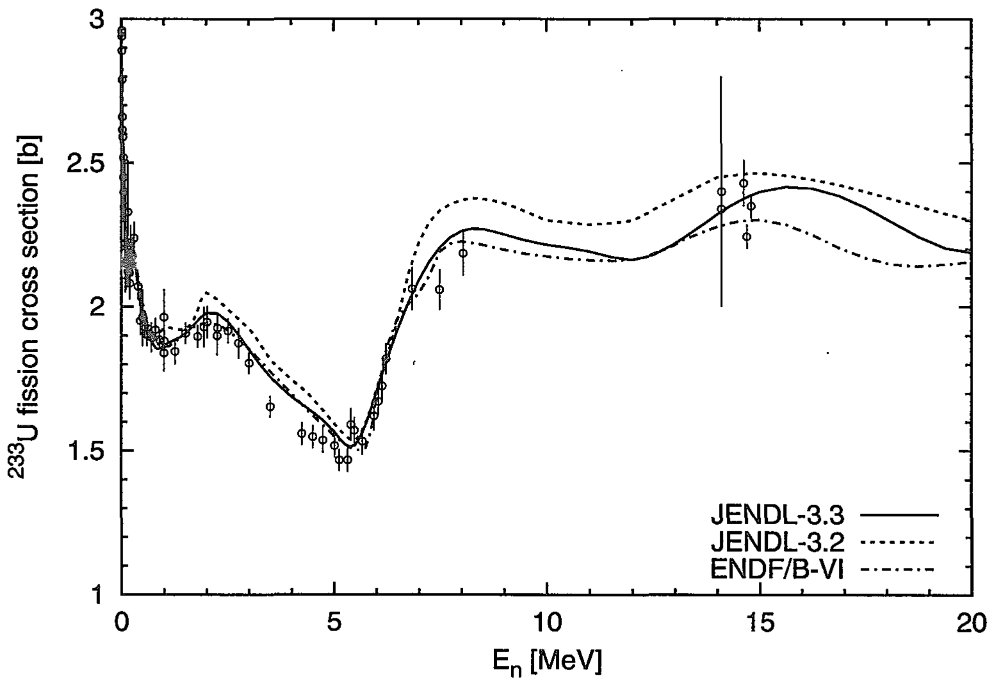

Fig. 4: Comparison of the fission cross sections of ^{233}U with the experimental data and the evaluated cross sections in JENDL-3.2 and ENDF/B-VI.

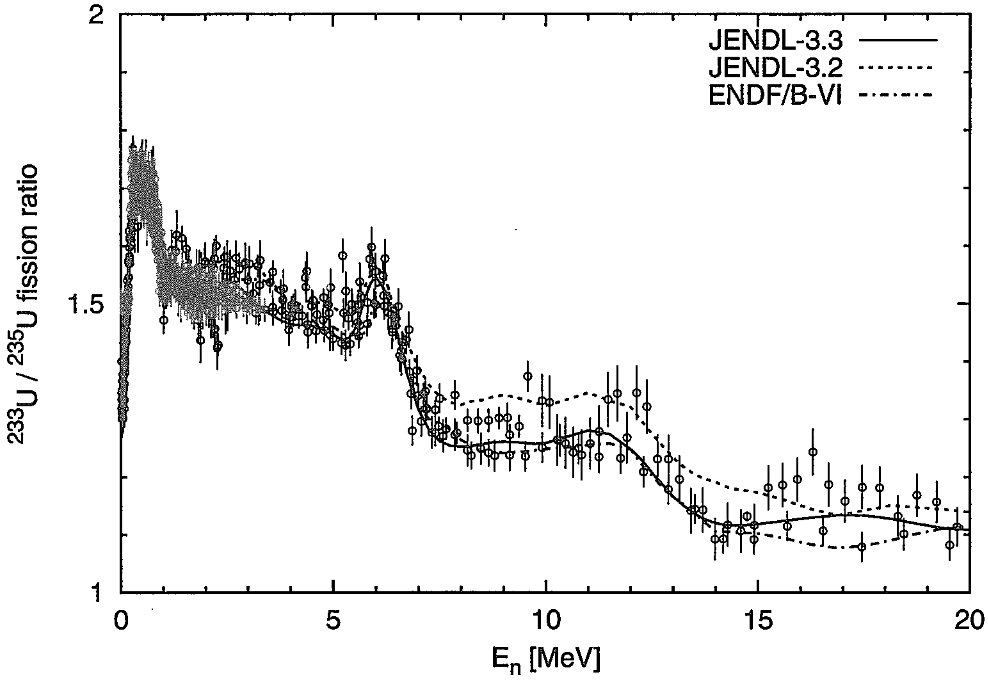

Fig. 5: Comparison of the fission cross section ratios of ^{233}U to ^{235}U with the experimental data, and with the evaluated values of JENDL-3.2 and ENDF/B-VI.

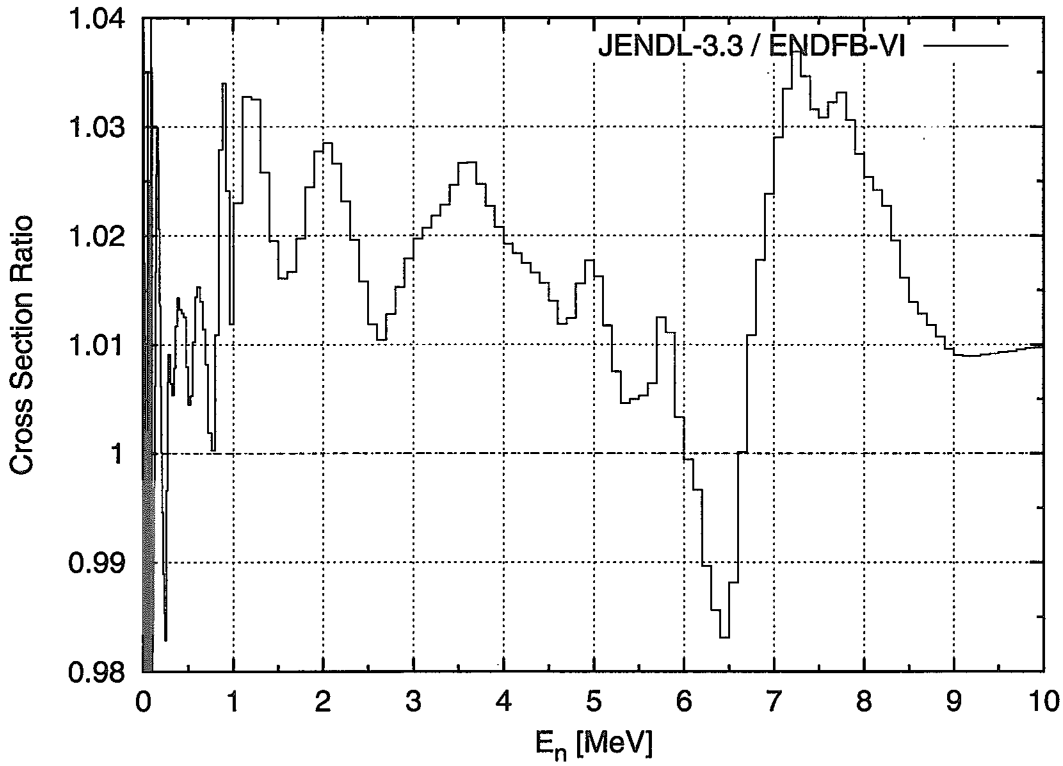

Fig. 6: Ratio of ^{235}U fission cross sections in JENDL-3.3 to those in ENDF/B-VI.

2.4 (n, xn) Cross Sections

The cross sections of $(n, 2n)$ and $(n, 3n)$ of ^{238}U were re-evaluated based on the experimental data. The SOK code[36] was used to evaluate cross sections and their covariances. The $(n, 4n)$ cross section was newly added to the library, though the cross section is very small. In Fig. 7 the evaluated cross sections of $(n, 2n)$ reaction are compared with the experimental data. The comparison for the $(n, 3n)$ reaction is shown in Fig. 8. The experimental data in those figures are Landrum *et al.*[41], Veeser and Arthur[42], Frehaut *et al.*[43], Karius *et al.*[44], Mather *et al.*[45], Batchelor *et al.*[46], Perkin and Coleman[47], Frehaut *et al.*[48], Ryves and Kolkowski[49], Chou[50], and Kornilov *et al.*[51]

In JENDL-3.2 an excitation function of the $(n, 2n)$ reaction of ^{235}U was distorted in order to conserve total reaction cross sections above 14 MeV, and this drawback was fixed in JENDL-3.3. A simple analytical function was fitted to the experimental data of Frehaut *et al.*[43] in the energy range from the threshold (5.32 MeV) to 13.09 MeV. Above 13.09 MeV the GNASH code was used to calculate the excitation function, and the calculated result was re-normalized to the experimental data (717 mb) at 13.09 MeV. The evaluated cross sections of $(n, 2n)$ reaction are compared with the experimental data of Frehaut *et al.*[48] in Fig. 9.

The $(n, 3n)$ and $(n, 4n)$ reaction cross sections of ^{235}U were also modified slightly as shown in Fig. 10, which is based on the experimental data of Veeser and Arthur[42].

Since the (n, xn) reaction data are inaccessible for ^{233}U, the GNASH calculations were used to evaluate the cross section. The GNASH results were re-normalized to the experimental fission-spectrum averaged cross section of Kobayashi *et al.*[52], which is $\overline{\sigma} = 4.08$ mb.

The evaluations of $(n, 2n)$, $(n, 3n)$, and $(n, 4n)$ reactions for ^{240}Pu and ^{242}Pu were made based on the statistical model calculation. The total neutron emission cross section was obtained by the coupled-channels calculation with ECIS[53], and this cross section was divided into various neutron emission channels. The branching ratios for each channel were taken from the calculation of Konshin[54]. The comparisons of those cross sections are shown in Figs. 11 and 12.

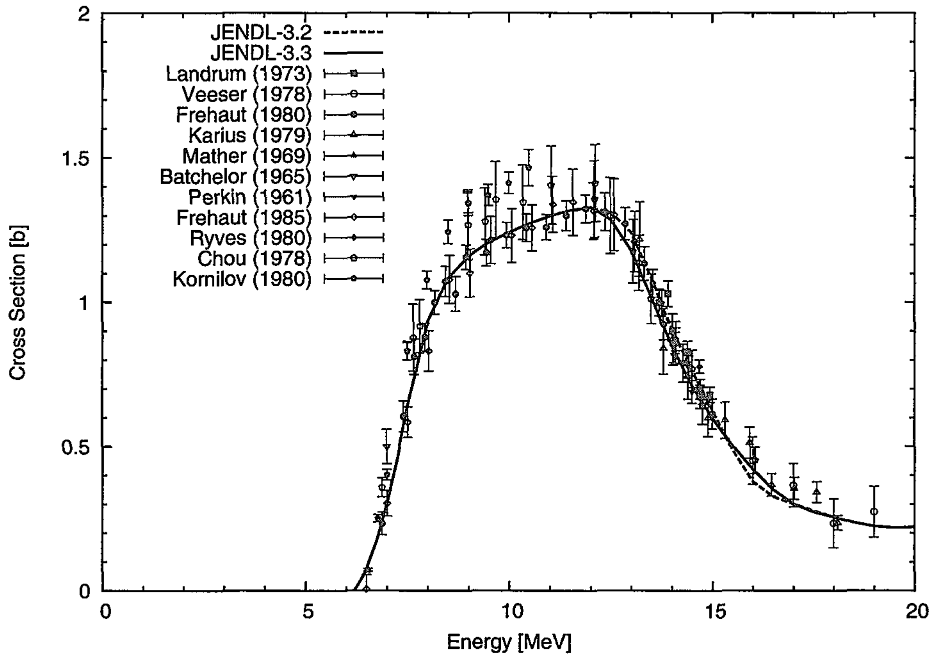

Fig. 7: Comparison of $(n, 2n)$ reaction cross section of ^{238}U in JENDL-3.2 and JENDL-3.3 with the experimental data.

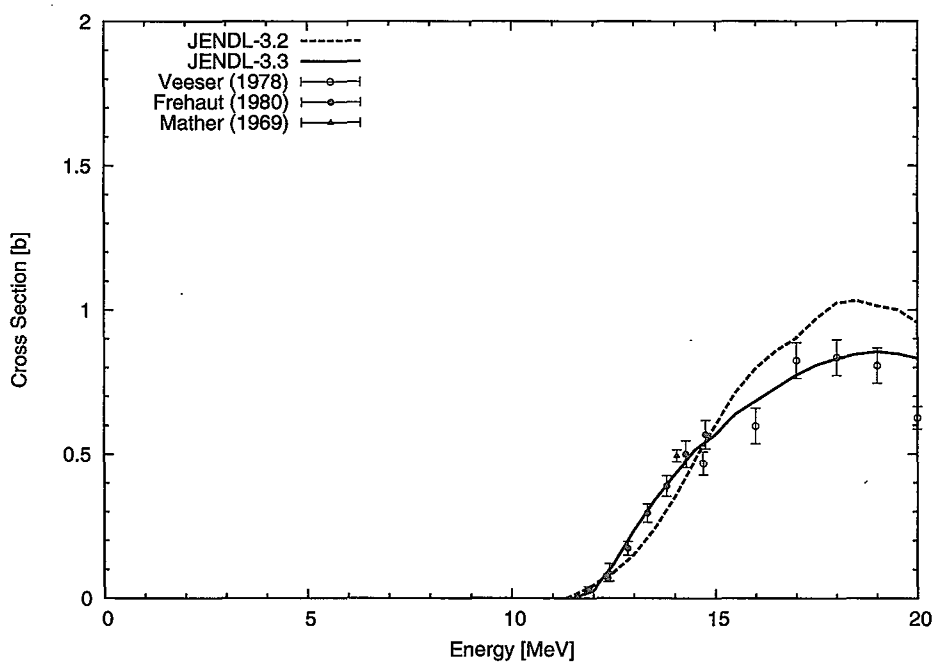

Fig. 8: Comparison of $(n, 3n)$ reaction cross section of ^{238}U in JENDL-3.2 and JENDL-3.3 with the experimental data.

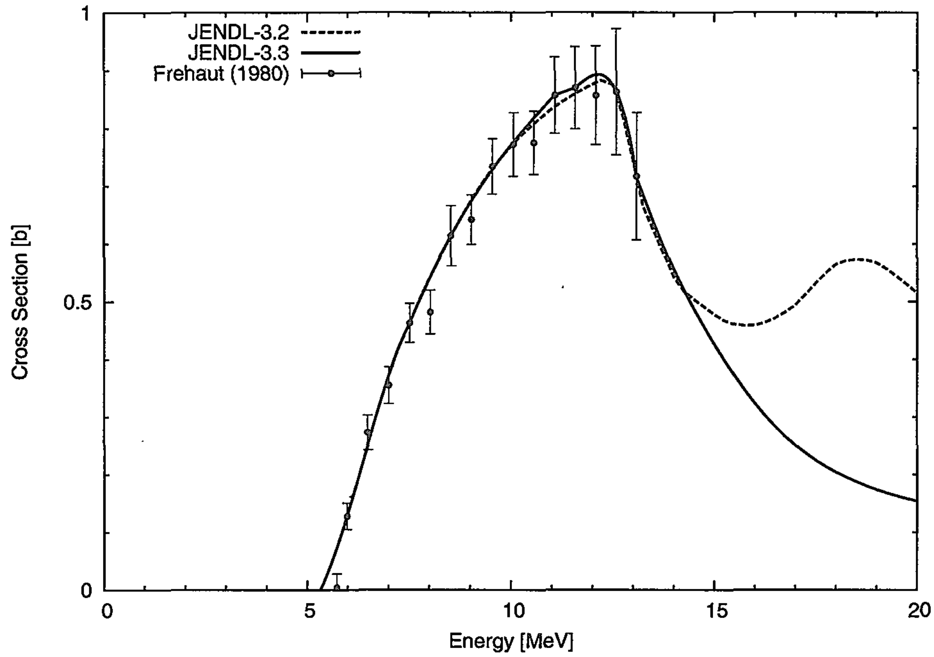

Fig. 9: Comparison of $(n, 2n)$ reaction cross section of ^{235}U in JENDL-3.2 and JENDL-3.3 with the experimental data.

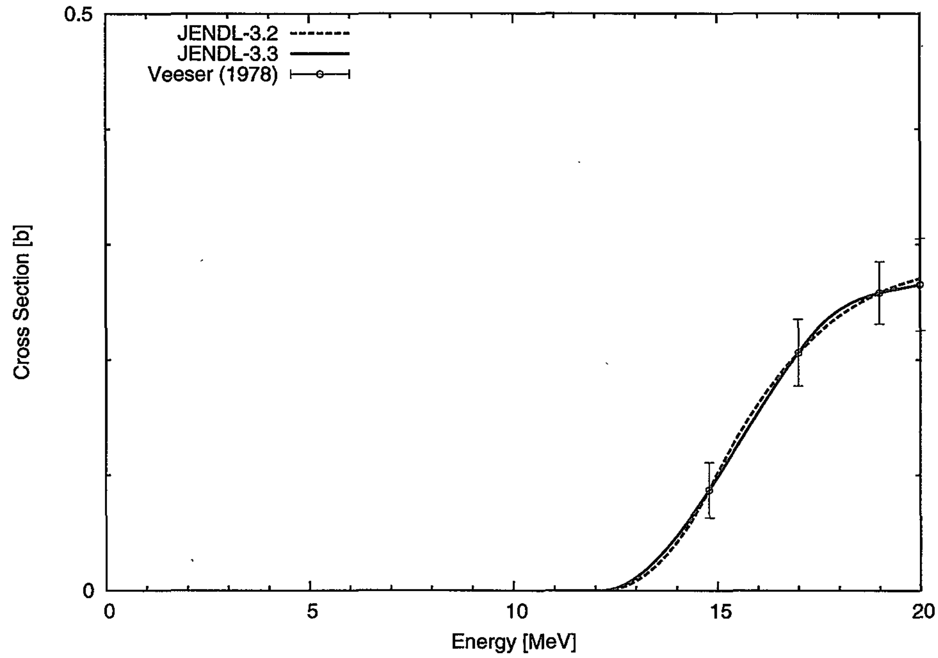

Fig. 10: Comparison of $(n, 3n)$ reaction cross section of ^{235}U in JENDL-3.2 and JENDL-3.3 with the experimental data.

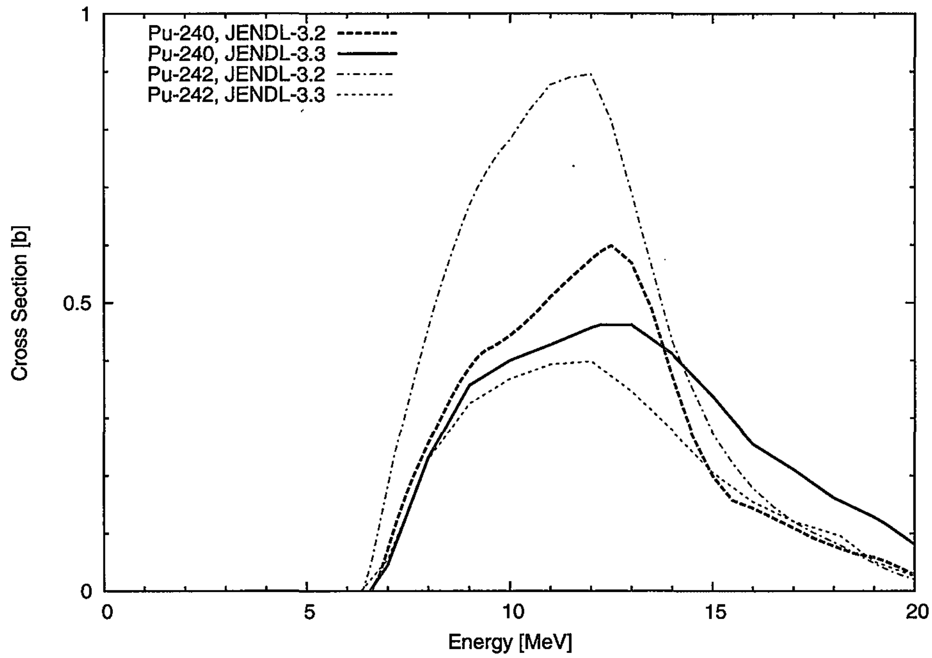

Fig. 11: Comparison of $(n, 2n)$ reaction cross section of ^{240}Pu and ^{242}Pu in JENDL-3.2 and JENDL-3.3.

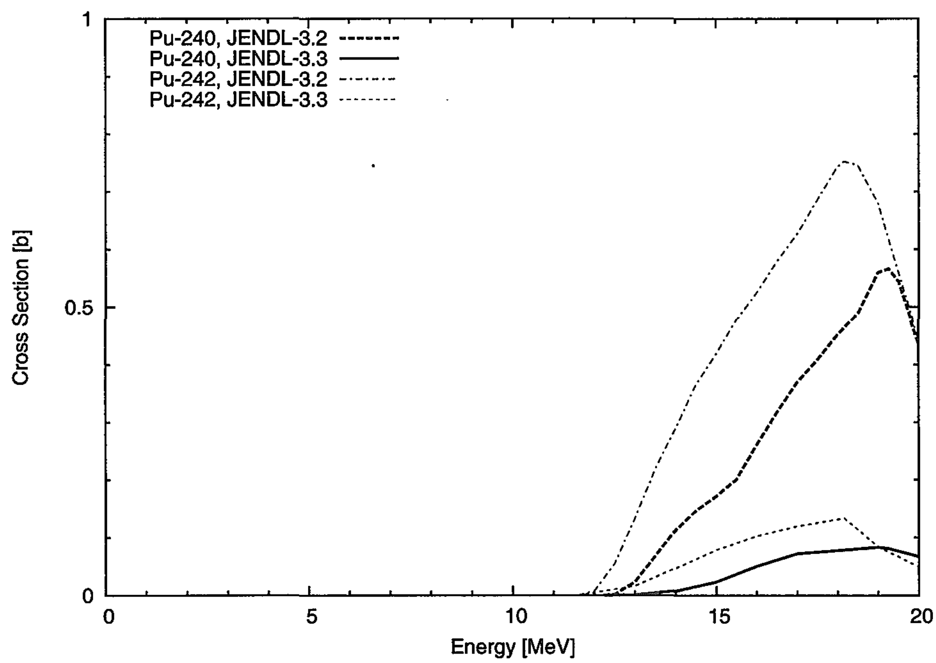

Fig. 12: Comparison of $(n, 3n)$ reaction cross section of ^{240}Pu and ^{242}Pu in JENDL-3.2 and JENDL-3.3.

2.5 Inelastic Scattering Cross Sections

In JENDL-3.2, ECIS88[53] was used to calculate a direct inelastic scattering process to the discrete levels of ^{238}U. The rotational, β- and γ-vibrational band levels were included in the coupled-channels calculation[55]. The strength adopted to calculate the cross sections of the octupole $K = 0^-$ band levels (680, 732, and 827 keV) was overestimated in JENDL-3.2, and the strength was decreased to reproduce the experimental data of Shao *et al.*[56] Figures 13, 14, and 15 show the comparisons of the evaluated cross sections with the experimental data.

The direct cross sections for the octupole vibrational states ($K = 0^-$ band, 597, 649, and 742 keV) of ^{240}Pu were also made with the ECIS88 code[53]. The calculated inelastic scattering cross sections to those excited levels are shown in Fig. 16. The same technique was also applied to ^{242}Pu.

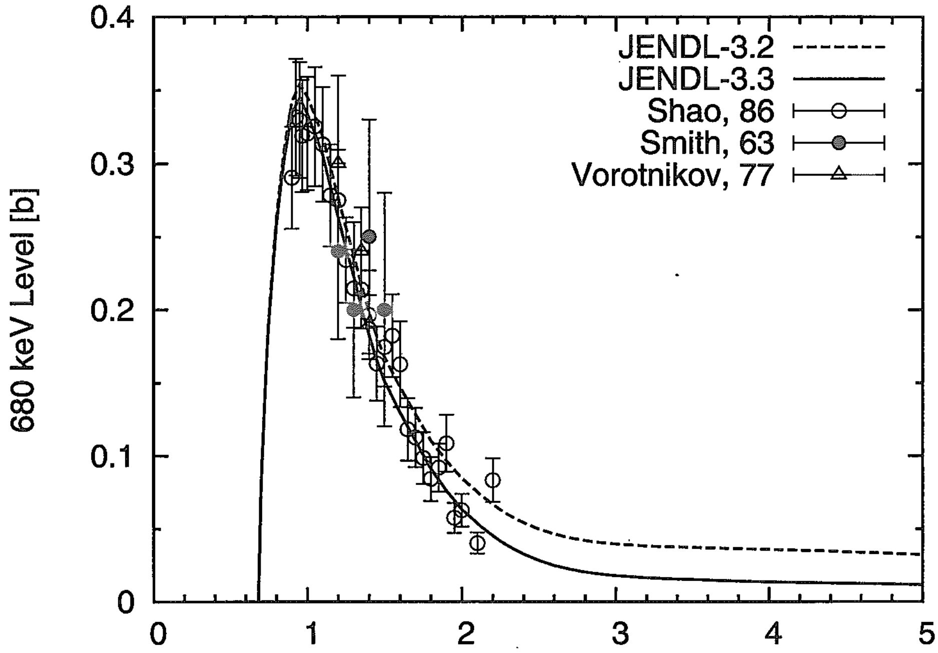

Fig. 13: Comparison of inelastic scattering cross section to the 680 keV level of ^{238}U in JENDL-3.2 and JENDL-3.3 with the experimental data.

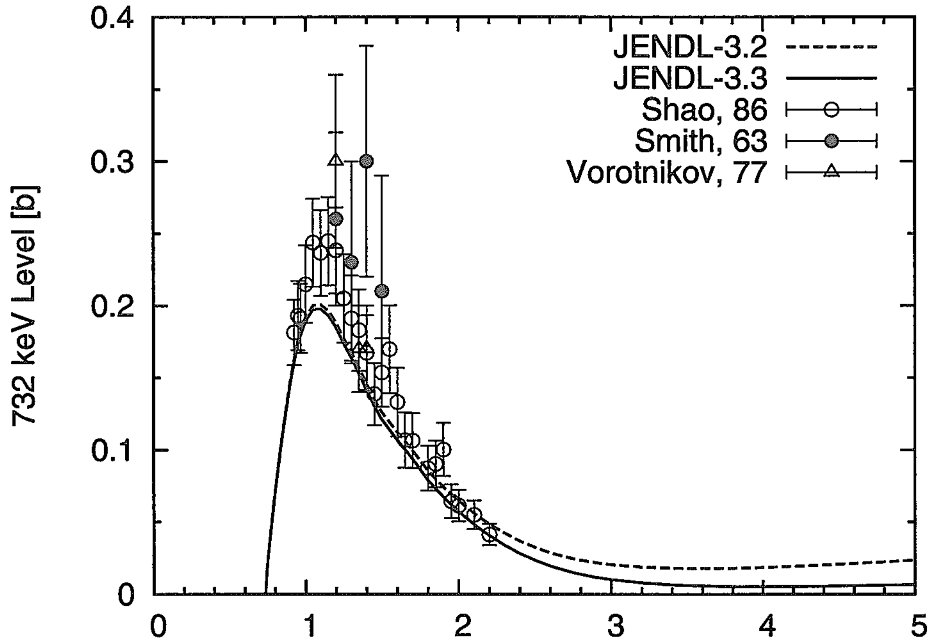

Fig. 14: Comparison of inelastic scattering cross section to the 732 keV level of ^{238}U in JENDL-3.2 and JENDL-3.3 with the experimental data.

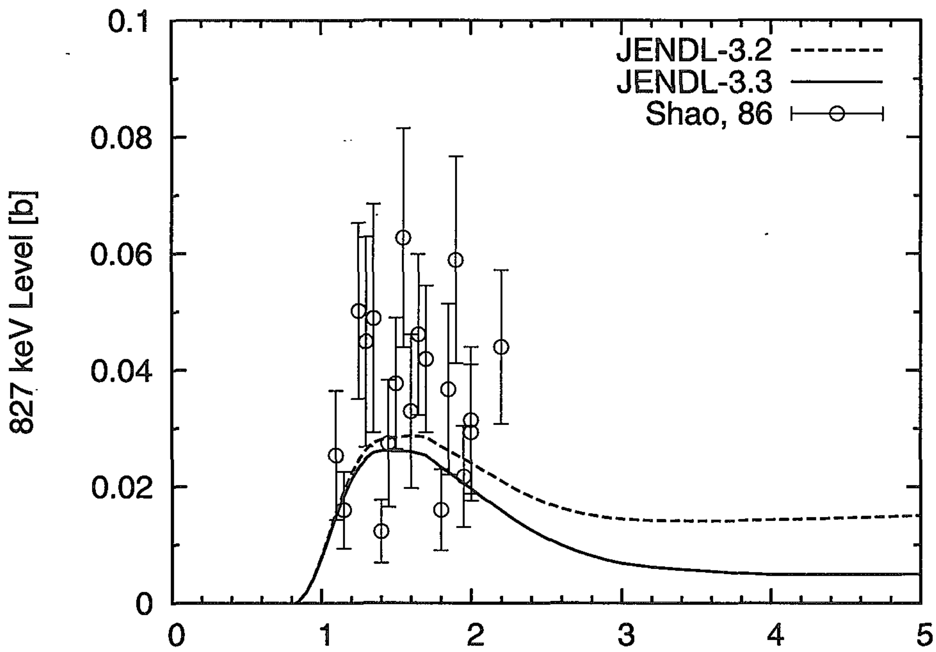

Fig. 15: Comparison of inelastic scattering cross section to the 827 keV levels of ^{238}U in JENDL-3.2 and JENDL-3.3 with the experimental data.

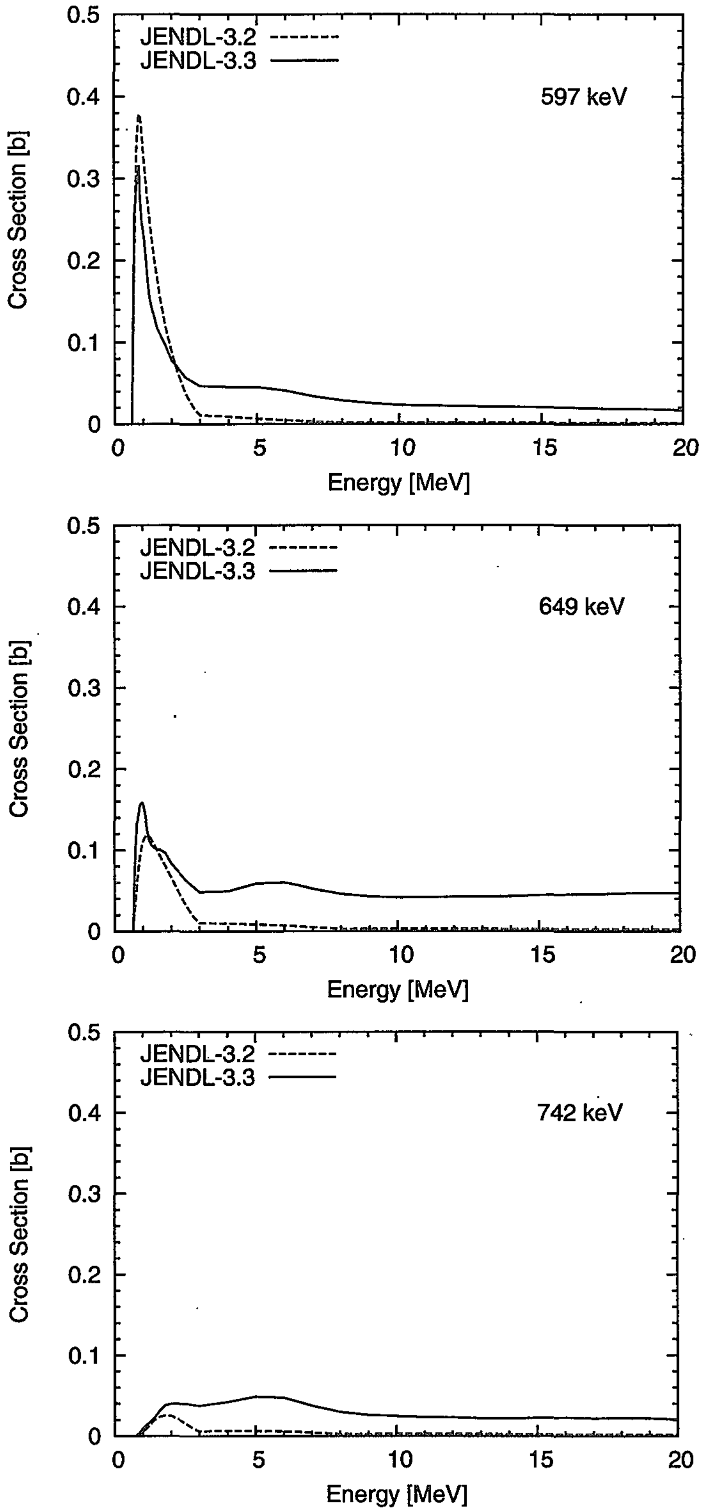

Fig. 16: Comparison of inelastic scattering cross sections to the 597, 649, and 742 keV levels of ^{240}Pu in JENDL-3.3 with those in JENDL-3.2.

2.6 Direct/Semidirect Capture Process

The radiative capture process above about 5 MeV is explained by the Direct/Semidirect (DSD) radiative capture theory[9, 10]. Since this process was not considered for many nuclei in JENDL-3.2, we made a computer program DSD[57] to calculate those cross sections. This code is based on the theory of Kitazawa *et al.*[58], which is the DSD theory for deformed nuclei, but we made some simplifications.

The Direct/Semidirect capture cross section is given by a coherent sum of the amplitudes for the direct and semidirect (collective) parts. The capture cross section for an incident wave (l_i, j_i) into a bound state (l, j) is

$$\sigma(l_i j_i; lj) = \frac{8\pi\mu}{9k\hbar^2}\left(\frac{E_\gamma}{\hbar c}\right)^3 |T^{\mathrm{D}} + T^{\mathrm{C}}|^2, \tag{2}$$

where E_γ is the γ-ray energy, k is the incident wave number, μ is the reduced mass, T^{D} is the amplitude for the direct capture, and T^{C} is for the semidirect capture. Those amplitudes include the radial matrix elements $\langle R_{ljK_f}(r)|r|R_{l_ij_i}(r)\rangle$ and $\langle R_{ljK_f}(r)|h(r)|R_{l_ij_i}(r)\rangle$, where R_{ljK} is the radial wave function of Nilsson model, $R_{l_ij_i}$ is the distorted wave calculated with an appropriate optical potential, and $h(r)$ is the particle-vibration coupling function. For our calculation we utilize the form $h(r) = V_1 r f(r)$ where $f(r)$ is the optical potential form factor, and V_1 is taken to be 110 MeV[58]. The Madland-Young optical potential[59] is used for $f(r)$, and the distorted wave is calculated with the same potential. The GDR (Giant-Dipole Resonance) parameters are taken from Ref. [58].

The maximum values of capture cross sections calculated with the DSD theory are in the order of 1 mb, which is too small in comparison with the other reaction channels such as elastic, inelastic, fission, *etc.* Such a small cross section is not so important for practical applications, then we made several approximations to make the calculation easy. The approximations are as follows: (i) the radial wave function $R_{ljK}(r)$ is calculated by assuming the target is spherical, (ii) the target is assumed to be doubly-closed shell nucleus, and (iii) the spherical optical potential is used to generate the scattering neutron wave function.

Equation (2) is calculated for all bound states above the Fermi energy. Due to the approximation (i) the calculated total capture cross section tends to overestimate the experimental data, we adjusted the number of bound states to which an incident neutron can be captured. This adjustment was made for ^{238}U since experimental data[60, 61] are available. Comparisons of the calculated capture cross sections with the experimental data[60, 61] are shown in Fig. 17. The dot-dashed line is calculated with the Hauser-Feshbach theory which predicts almost negligible cross sections above 5 MeV because the neutron width Γ_n is much larger than the capture width Γ_γ when many inelastic channels open. A sum of the Hauser-Feshbach and DSD calculations gives the final evaluated cross sections in this energy range.

Although we made several approximations to the original DSD model, the calculated values are very similar to those by McDaniels *et al.* and Kitazawa *et al.* (see Fig. 2 in Ref. [60]), therefore our simplified version of DSD model is not so crucial, and can be applied for other nuclides. Since experimental data for the capture cross section are inaccessible for many nuclei at the energies above 10 MeV, we estimated the cross section with the same code but the parameters are selected appropriately. The optical potential of Madland and Young was used for many actinides. The GDR parameters for each nuclide were taken from RIPL[62] (Reference Input Parameter Library).

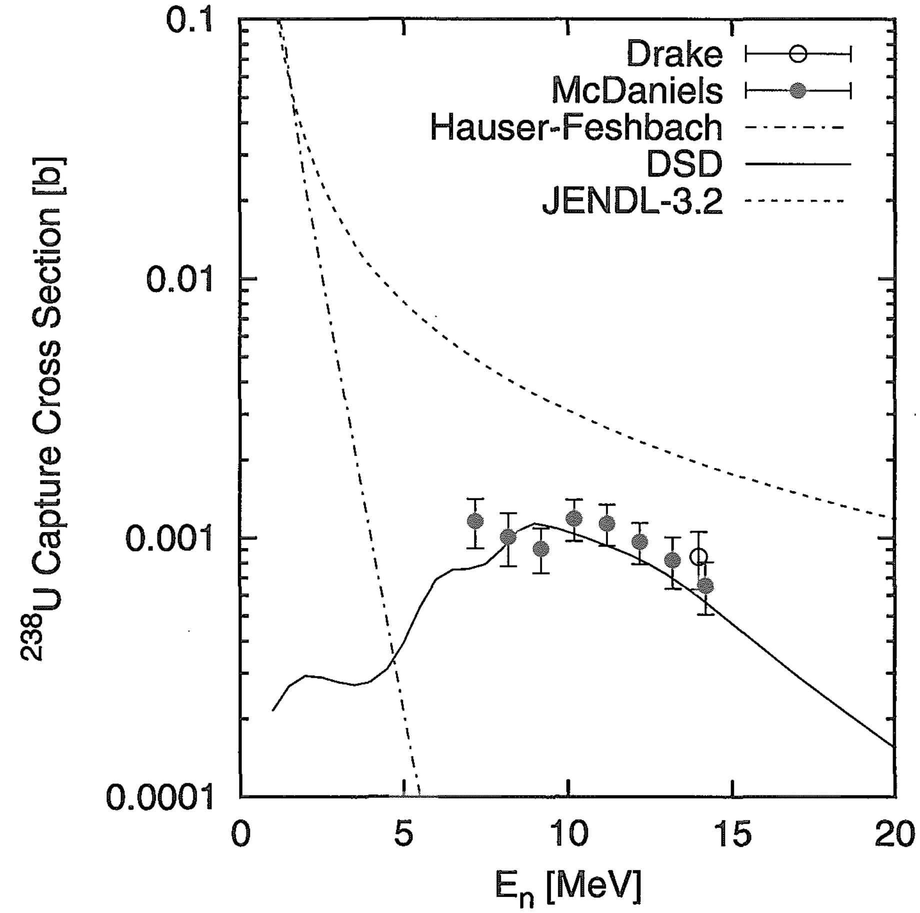

Fig. 17: Direct/Semidirect model calculation for ^{238}U capture cross sections. The dot-dashed line is calculated cross sections with the Hauser-Feshbach theory, and the solid line is the result of the DSD model calculation. The dotted line shows the cross sections in JENDL-3.2, which were evaluated empirically.

2.7 Prompt Fission Neutron Spectrum

2.7.1 Multimodal Fission Analysis

The prompt fission neutron spectra for ^{235}U and ^{239}Pu were re-evaluated by Ohsawa[63] with a multimodal fission analysis[5, 6, 64]. The fission neutron spectrum is decomposed into four components, each of which corresponds to a fission mode. Those are Standard 1, 2, 3, and Super Long modes. The prompt fission neutron spectra for those modes were calculated individually by the Madland-Nix model[65, 66] with some modifications by Ohsawa and co-workers[67, 68, 69].

Ohsawa's model allows the weight of two fission fragments to be different, and the fission spectrum is calculated as a weighted average of spectra from both the light and heavy fragments,

$$\chi(E) = \left(\nu_L \chi_L(E) + \nu_H \chi_H(E)\right) / (\nu_L + \nu_H), \tag{3}$$

where χ_L and χ_H are the normalized spectra corresponding to each fragment, and ν the average number of neutrons. The ratio of ν_L to ν_H is not well known at high energies. In such a case $\nu_L = \nu_H$ is assumed.

The spectrum $\chi_{L,H}$ is given by [65]

$$\chi_{L,H}(E) = \frac{1}{2\sqrt{E_f T_m^2}} \int_{(\sqrt{E}-\sqrt{E_f})^2}^{(\sqrt{E}+\sqrt{E_f})^2} \sigma_R(\epsilon)\sqrt{\epsilon}d\epsilon \int_0^{T_m} k(T) T \exp^{-\epsilon/T} dT, \tag{4}$$

where E_f and T_m are the average energy and the maximum nuclear temperature of the fission fragments, $k(T)$ the temperature dependent normalization integral [65], and $\sigma_R(\epsilon)$ the inverse reaction cross section calculated with the optical model. The maximum temperature T_m can be related to the neutron binding energy B_n, the incident energy E_n, the total energy release E_r, and the total kinetic energy E_k as

$$aT_m^2 = E_r + B_n + E_n - E_k, \tag{5}$$

where a is the level density parameter of the compound nucleus. The total energy release is calculated with the mass formula of Tachibana *et al.* [70], and the values of E_k are taken from measurements. The temperatures for the light and heavy fragments are determined from the relation

$$a_L T_{mL}^2 + a_H T_{mH}^2 = aT_m^2. \tag{6}$$

The level density formula of Ignatyuk *et al.* [71] is adopted for the fission fragments in order to include the shell effect.

In the multimode analysis Eq. (3) is calculated for each mode namely Standard 1 (S1), Standard 2 (S2), Standard 3 (S3), and Super Long (SL) modes. Those are averaged to give experimentally observable fission spectra.

$$\overline{\chi}(E) = \frac{\sum_i w_i \nu_i \chi_i(E)}{\sum_i w_i \nu_i}, \quad i = \text{S1, S2, S3, and SL} \tag{7}$$

where $\chi_i(E)$ stands for the fission spectrum for each mode, and w_i is the mode branching ratio. The mode branching ratios used are show in Tables 8 and 9. The calculated fission spectrum of ^{235}U at the thermal energy is shown in Fig. 18.

The fission spectrum for ^{235}U obtained at the thermal energy became harder than that in JENDL-3.2, but still slightly softer than that in ENDF/B-VI. We examined the new fission spectrum by calculating 25 fission spectrum averaged cross sections with evaluated values by Mannhart[72]. Those reactions are ^{19}F$(n,2n)$, ^{24}Mg(n,p), ^{27}Al(n,p) and (n,α), ^{32}S(n,p), ^{46}Ti(n,p), ^{47}Ti(n,p), ^{48}Ti(n,p), ^{55}Mn$(n,2n)$, ^{54}Fe(n,p), ^{56}Fe(n,p), ^{59}Co(n,α) and $(n,2n)$, ^{58}Ni(n,p) and $(n,2n)$, ^{63}Cu(n,α) and $(n,2n)$, ^{64}Zn(n,p), ^{90}Zr$(n,2n)$, ^{115}In(n,n'), ^{127}I$(n,2n)$, ^{197}Au$(n,2n)$, ^{235}U(n,f), ^{238}U(n,f), and ^{239}Pu(n,f). The dosimetry cross sections were taken from JENDL Dosimetry File 99[73]. Figure 19 shows the C/E values for those 25 dosimetry reactions. Three kinds of calculated integral cross sections are shown in this figure — namely those with the fission spectrum in JENDL-3.2, JENDL-3.3, and ENDF/B-VI. We found that the calculated fission spectrum averaged cross sections with the new spectrum agree very well with the Mannhart's evaluations. The averaged C/E values for those three evaluations are 0.9014 for JENDL-3.2, 0.9630 for JENDL-3.3, and 0.9789 for ENDF/B-VI.

With this hard fission spectrum, several benchmark tests for thermal reactors were carried out[74]. They reported that the overestimation of $k_{\rm eff}$ can be reduced by the hard fission spectrum and the spectrum in JENDL-3.3 is favorable for reactor calculations.

The multimode analysis was adopted for ^{235}U and ^{239}Pu up to the incident neutron energy of 5 MeV[63]. The fission spectra for ^{238}U in the energy range $0 \sim 5$ MeV are the same as those in JENDL-3.2.

Table 8: Parameters w_i and ν_i used for calculations of prompt fission neutron spectra for ^{235}U.

	fission mode			
	S1	S2	S3	SL
ν_L	1.0487	1.4672	0.0	3.1409
ν_H	1.3139	1.1392	0.0	2.5821
w	0.1834	0.8159	0.0	0.0007

Table 9: Parameters w_i and ν_i used for calculations of prompt fission neutron spectra for ^{239}Pu.

	fission mode			
	S1	S2	S3	SL
ν_L	1.58	1.40	0.32	0.0
ν_H	0.86	1.88	2.58	0.0
w	0.248	0.742	0.01	0.0

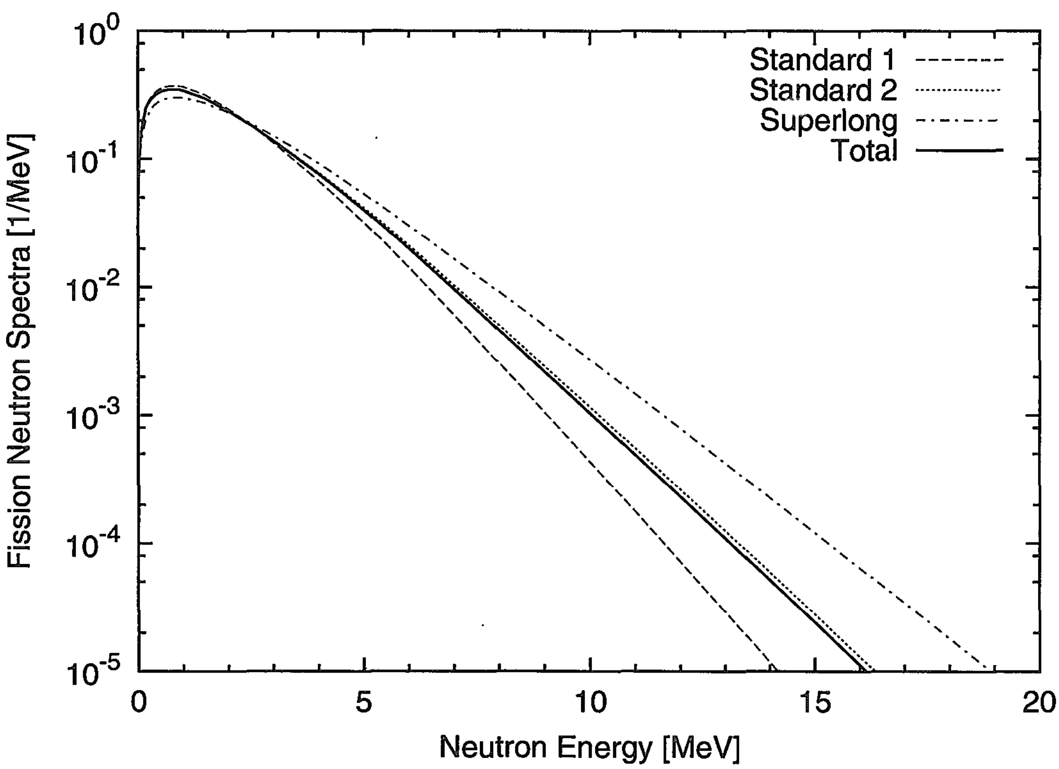

Fig. 18: Prompt fission neutron spectum of ^{235}U at thermal energy. The dashed line is the spectrum from the Standard 1 mode, the dotted line is the Standard 2 mode, and the dot-dashed line is the Superlong mode, respectively.

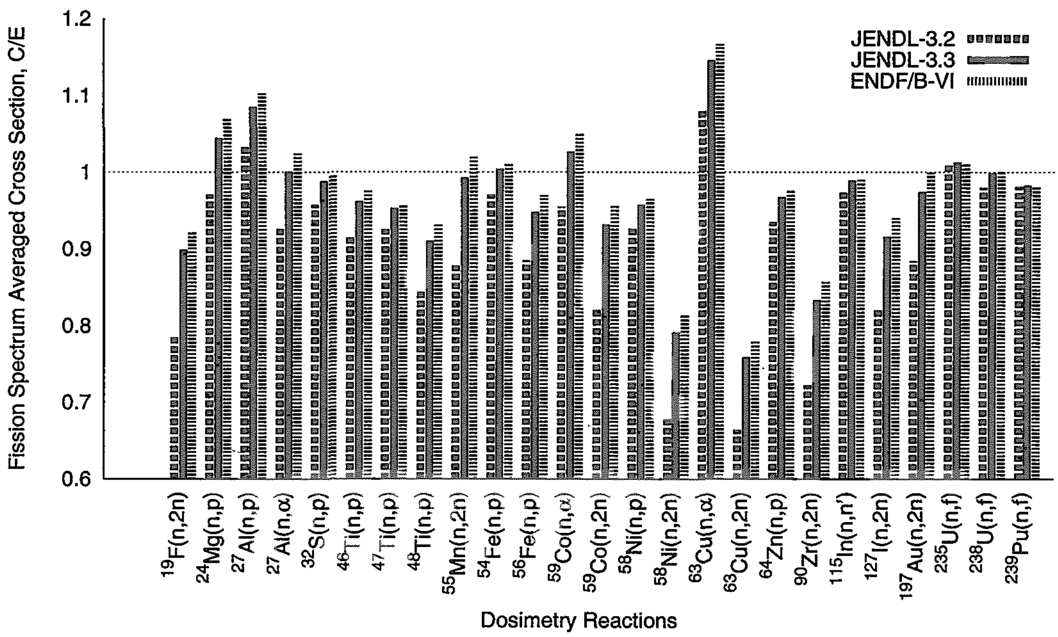

Fig. 19: C/E values of integral cross sections calculated with the prompt fission neutron spectra in JENDL-3.2, JENDL-3.3, and ENDF/B-VI.

2.7.2 Effect of Prefission Neutron

When an incident neutron energy is higher than a threshold energy of the $(n, 2n)$ reaction, a multiple-chance fission occurs. This effect was already taken into account in the JENDL-3.2 evaluation, however recent findings for the multiple-chance fission were an effect of preequilibrium process on a prefission neutron which is emitted before scission. This effect was calculated[7] for ^{235}U, ^{238}U, and ^{239}Pu with the FKK theory[75] and compiled into the JENDL-3.3 evaluation. Parameters for the FKK calculation were determined by the ^{238}U experimental data[76, 77], and the same parameters were used for the ^{235}U and ^{239}Pu calculations.

The fission spectra for those multiple-chance fissions can be obtained by a weighted sum of the spectra from fission fragments and the spectra from a statistical decay before scission. The weight is calculated with the average number of neutrons, ν_i, and the fission probability P_{fi}, where i stands for the ith chance fission. The fission probabilities were taken from experimental data.

The first-, second-, and third-chance fission neutrons excluding the prefission neutrons are expressed by $\nu_1\sigma_{1f}\chi_1(E)$, $\nu_2\sigma_{nf}\chi_2(E)$, and $\nu_3\sigma_{2nf}\chi_3(E)$, where $\chi_i(E)$ is calculated by Eq. (4). The multiple-chance fission cross sections σ_{1f}, σ_{nf}, and σ_{2nf} can be obtained by a decomposition of the total fission cross section σ_f. Note that the first-chance fission cross section is denoted by σ_{1f} in order to distinguish from the total fission cross section σ_f.

The inelastic scattering neutron $(n, n'X)$ has a normalized spectrum $\phi_1(E)$. Since it is an inclusive process, the cross section σ_{nX} is given by a sum of all possible neutron emission reactions; then, $\sigma_{nX} = \sigma_{n'} + \sigma_{2n} + \sigma_{3n} + \sigma_{nf} + \sigma_{2nf}$. The spectrum $\phi_1(E)$ has a forward-peaked angular distribution, and it is calculated with the FKK model[75]. For the $(n, 2nX)$ and $(n, 3nX)$ reactions, two or three neutrons are evaporated from the compound before scission. Those spectra are expressed as $\sigma_{2nX}\phi_2(E)$ and $\sigma_{3n}\phi_3(E)$, where $\sigma_{2nX} = \sigma_{2n} + \sigma_{3n} + \sigma_{2nf}$. The spectra $\phi_2(E)$ and $\phi_3(E)$ are assumed to be isotropic in the center-of-mass system, and those are calculated with the Hauser-Feshbach theory. With the quantities defined above, the observable angle-integrated cross sections can be represented by

$$\frac{d\sigma}{dE} = \sigma_{nX}\phi_1(E) + \sigma_{2nX}\phi_2(E) + \sigma_{3n}\phi_3(E) + \nu_1\sigma_{1f}\chi_1(E) + \nu_2\sigma_{nf}\chi_2(E) + \nu_3\sigma_{2nf}\chi_3(E). \quad (8)$$

The total energy spectra in Eq. (8) contain contributions of the reactions in which fission does not take place, such as (n, n') and $(n, 2n)$, therefore those processes should be excluded. We subtracted a contribution of the (n, n') reaction in which the excitation energy of the residual nucleus is lower than the fission barrier energy. On the other hand, we did not pay attention to the spectra from $(n, 2n)$ and $(n, 3n)$, because those contributions were small.

Normalized fission spectrum of ^{238}U at neutron-induced energy of 18 MeV is shown in Fig. 20. The preequilibrium effect can be observed in the secondary neutron energy region 5–12 MeV. Above 12 MeV the fission neutron spectrum does not contain the prefission component, so that the preequilibrium effect is sharply cut off there, and its component is observed as inelastically scattered neutrons. Because of the existence of the prefission neutron, the angle-differential spectra become anisotropic in the energy region 5–12 MeV.

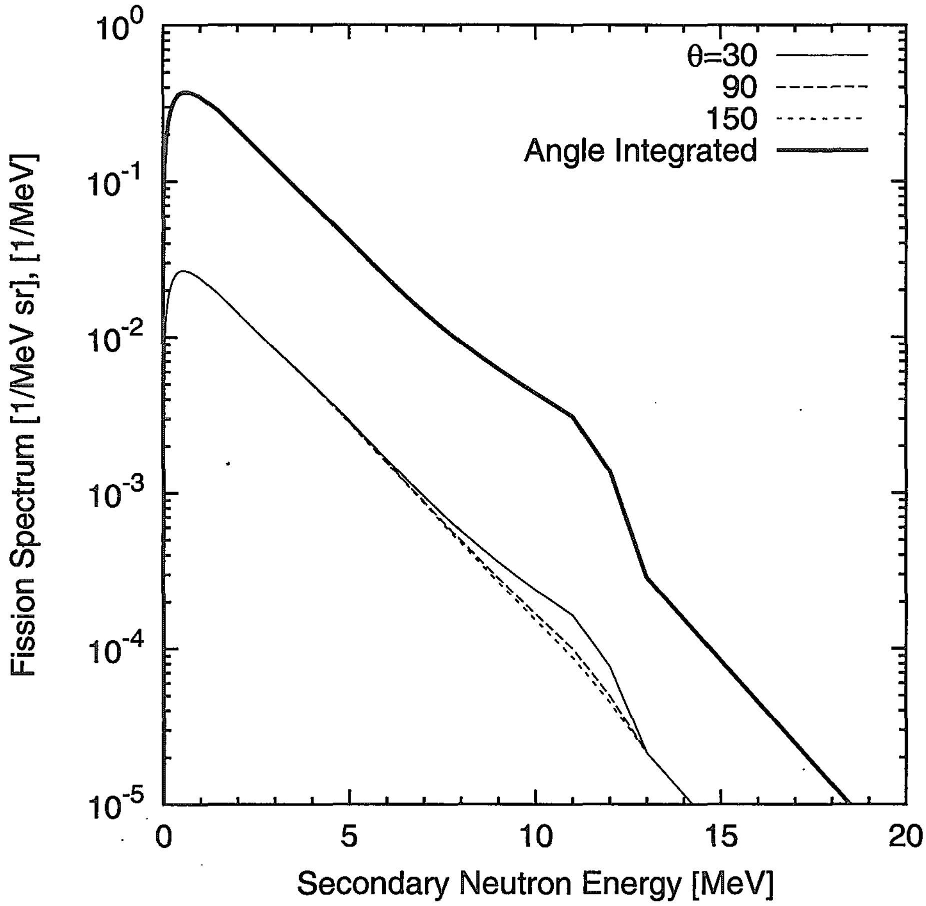

Fig. 20: Normalized prompt-neutron fission neutron spectra of ^{238}U at the neutron incident energy of 18 MeV. The double-differential fission spectra have a unit of (MeV sr)$^{-1}$, while the unit of angle-integrated spectrum is MeV^{-1}.

2.8 Secondary Neutron Energy Spectrum

Secondary neutron energy spectra for the (n, n'), $(n, 2n)$, and $(n, 3n)$ reactions compiled in JENDL-3.2 have been criticized for a long time. There exist three reasons for this problem. The first one is an adoption of the evaporation formula for neutron energy spectra above a threshold energy of $(n, 2n)$ reaction. This problem arises for several minor actinides. The second one is a special case for ^{238}U. The energy spectra of ^{238}U were calculated with the GNASH code[4], and the crude output was stored into JENDL-3.2 without any post-processing procedures. The third one is not crucial. In JENDL-3.2, energy spectra for many important nuclei were calculated with the PEGASUS code[78]. This is an evaporation and preequilibrium model calculation code and generates an appropriate spectrum in the ENDF format, although the use of the GNASH code may be preferable in view of more accurate evaluation.

In the present revision work, we adopted the GNASH code, and the results were processed with the GAMFIL code[79]. With this procedure the crucial problems concerning the energy spectra were fixed. Figures 21, 22, and 23 show comparisons of energy spectra for the (n, n'), $(n, 2n)$, and $(n, 3n)$ reactions on ^{235}U in JENDL-3.2 and JENDL-3.3. Each line in these figures corresponds to a neutron incident energy.

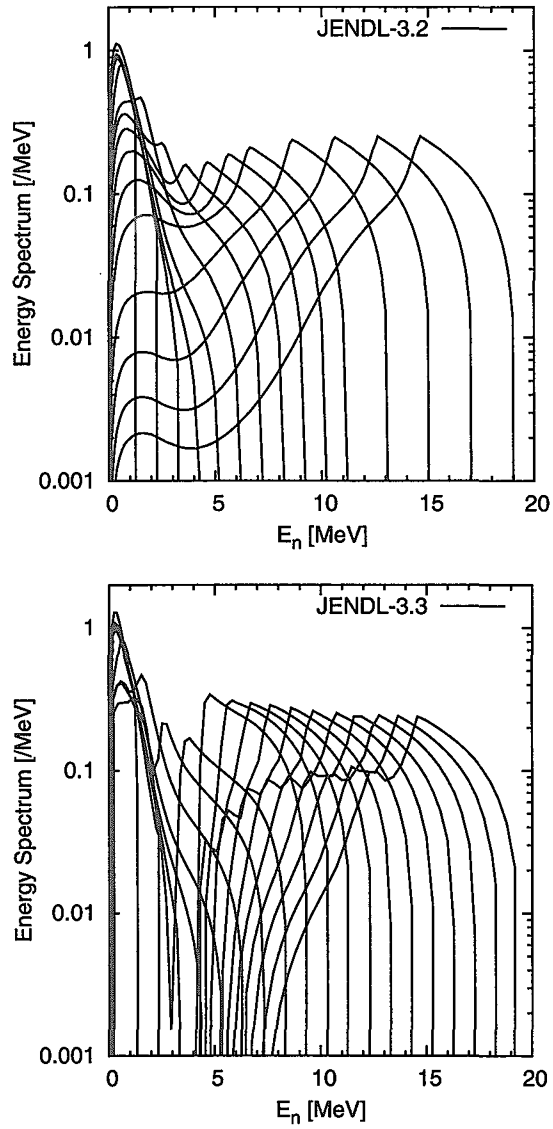

Fig. 21: Secondary neutron energy spectra for the (n, n') reaction on ^{235}U. Each line corresponds to a neutron incident energy.

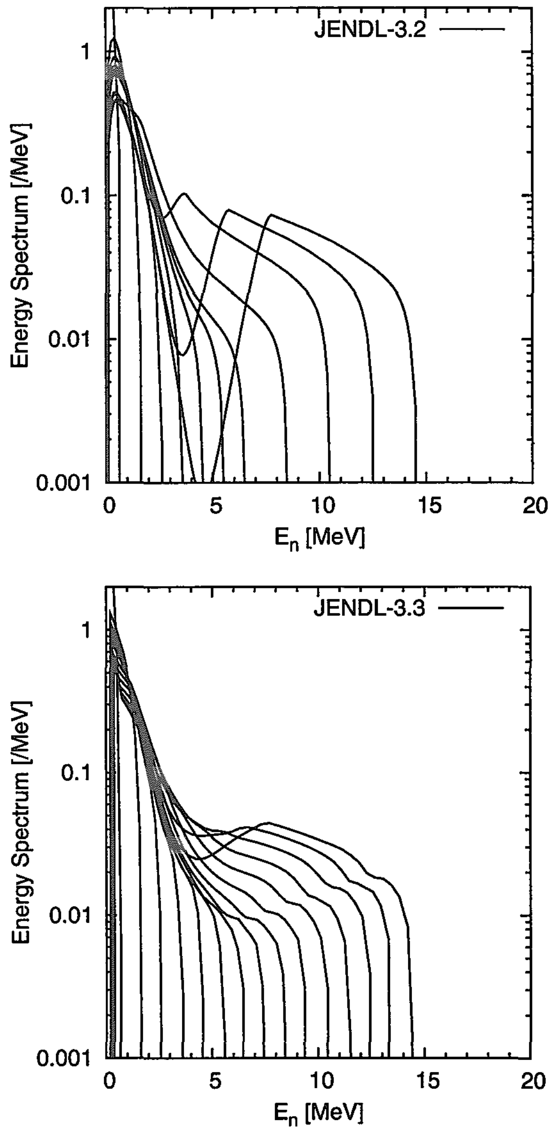

Fig. 22: Secondary neutron energy spectra for the $(n, 2n)$ reaction on ^{235}U.

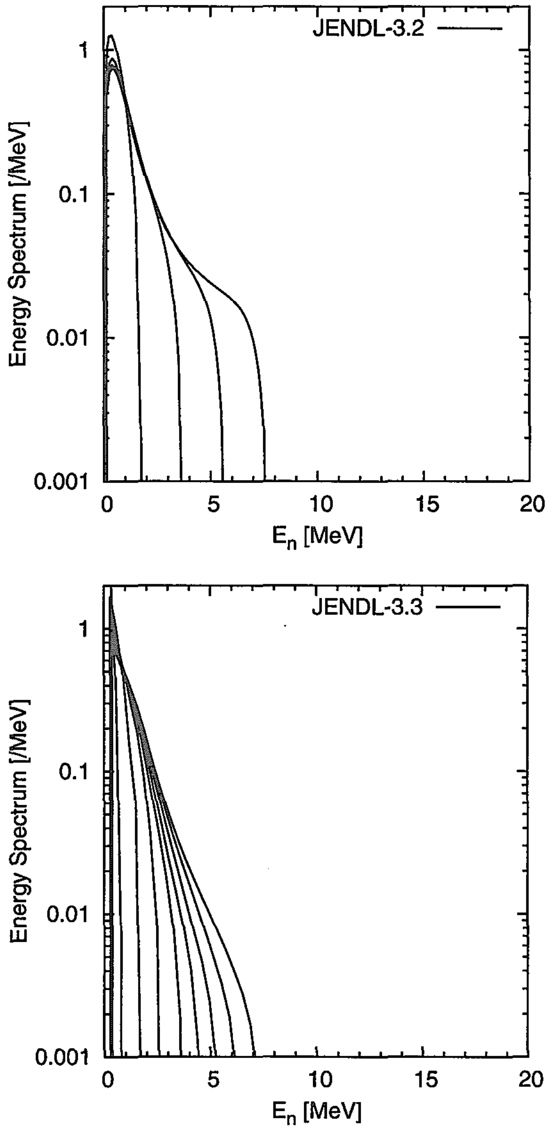

Fig. 23: Secondary neutron energy spectra for the $(n, 3n)$ reaction on ^{235}U.

2.9 Number of Prompt Neutrons per Fission

The prompt ν's of ^{233}U and ^{235}U were re-evaluated by fitting simple functions to the experimental data available. The GMA code[39] was used to evaluate those covariances.

The experimental data used for the evaluation of ν_p value of ^{235}U are, Gwin *et al.*[80, 81, 82, 83], Frehaut *et al.*[84, 85, 86], and Howe[87]. Figure 24 shows the comparison of ν_p for ^{235}U. The experimental data of Gwin *et al.*[80, 81] are the ratio to the neutron emission from spontaneous fission of ^{252}Cf. We adopted $\overline{\nu} = 3.756$ for ^{252}Cf spontaneous fission.

For ^{233}U, the adopted experimental data are, Gwin *et al.*[80, 81, 88], Reed *et al.*[89], Nurpeisov *et al.*[90, 91], Sergachev *et al.*[92], Boldeman and Walsh[93], Mather *et al.*[94], Protopopov and Blinov[95], Smirenkin *et al.*[96]. The comparisons of ν_p for ^{233}U in JENDL-3.2 and JENDL-3.3 are shown in Fig. 25.

Below 100 keV the difference between those two evaluations are very small. At the thermal energy, ν_p's of ^{233}U and ^{235}U in JENDL-3.2 and 3.3 are as follows:

	ν_p of ^{233}U	ν_p of ^{235}U
JENDL-3.2	2.48600	2.42038
JENDL-3.3	2.48098	2.42048
ENDF/B-VI	2.48730	2.42000

Difference between ν_p's of ^{233}U in JENDL-3.3 and ENDF/B-VI is about 0.25%, while the difference among those evaluations is almost negligible (0.02%) in the case of ^{235}U.

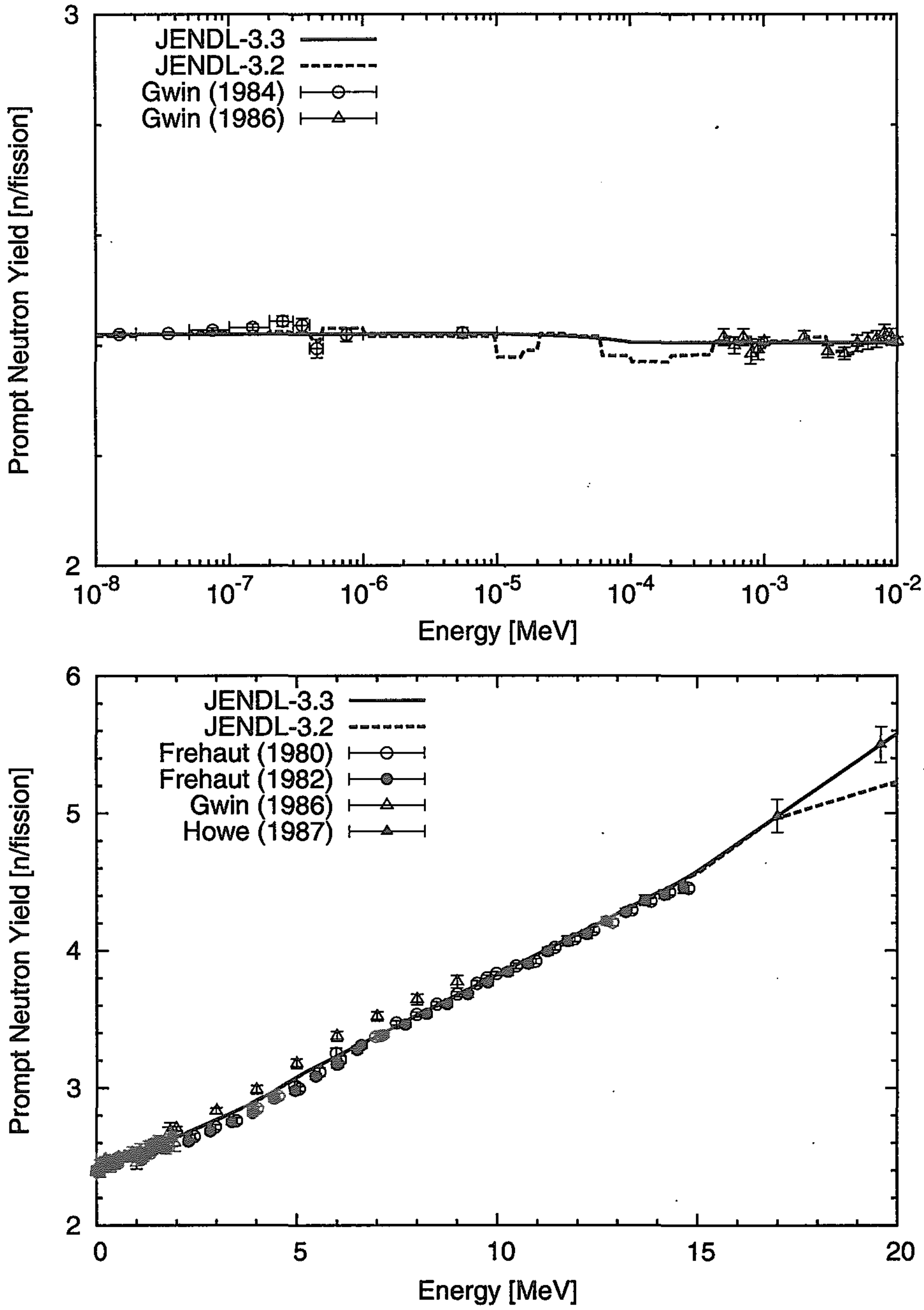

Fig. 24: Comparison of ν_p of ^{235}U in JENDL-3.2 and JENDL-3.3. The experimental data of Gwin *et al.*[80, 81] were obtained by assuming $\overline{\nu} = 3.756$ for ^{252}Cf spontaneous fission.

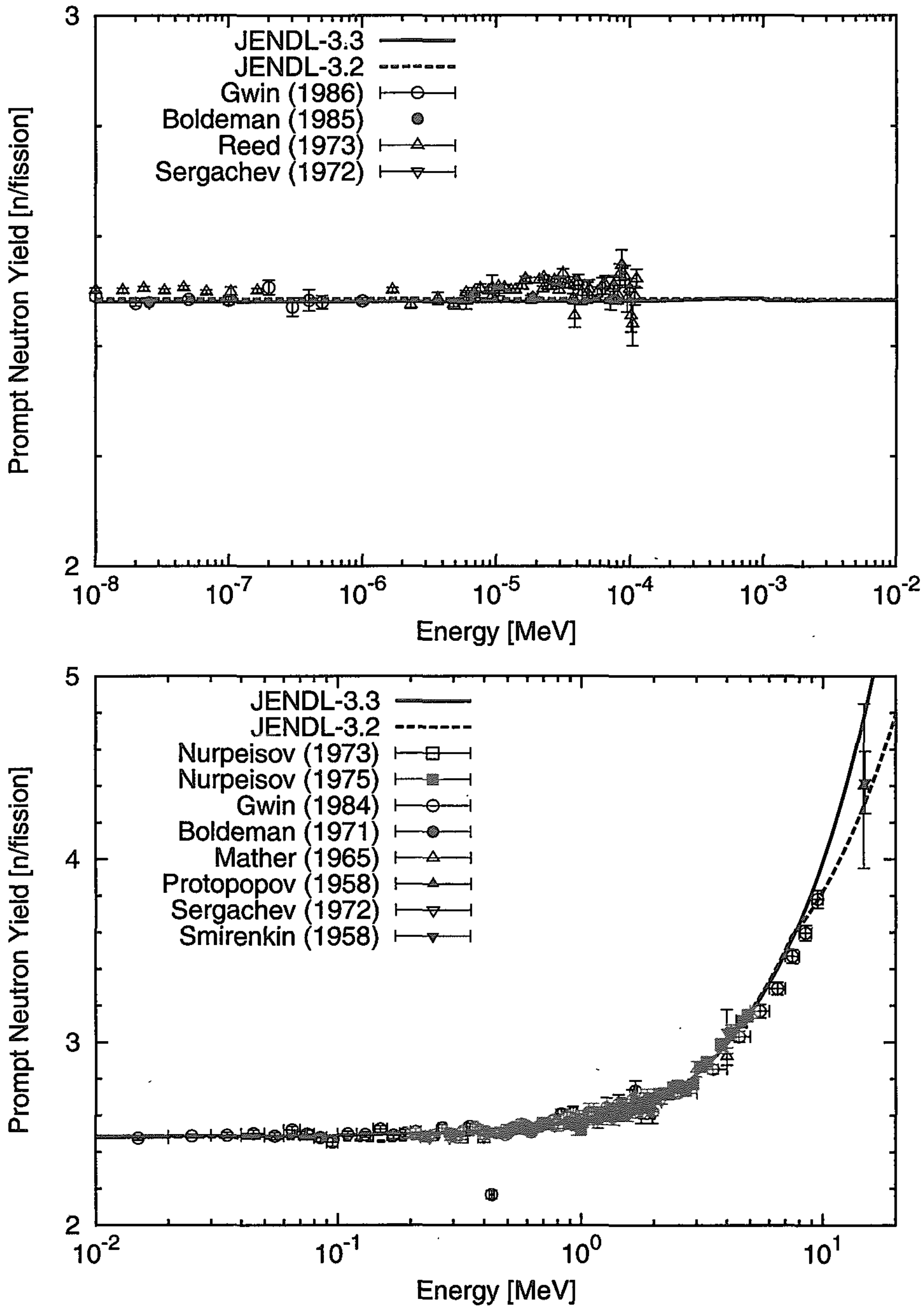

Fig. 25: Comparison of ν_p of ^{233}U in JENDL-3.2 and JENDL-3.3

2.10 Number of Delayed Neutrons per Fission

Revision of the number of delayed neutron per fission, ν_d, was carried out in collaboration with the Delayed Neutron Working Group[97] in JNDC. They investigated β_{eff} measurements at FCA, MASURCA, and TCA, and obtained a recommendation of ν_d values for ^{235}U, ^{238}U, and ^{239}Pu by an adjustment of those values in JENDL-3.2 to the integral measurements. Since the adjustment is only feasible in the thermal and epithermal energy regions, we re-evaluated the ν_d values for those nuclides in the whole energy region with the help of their adjustment results. Those re-evaluations were basically guided by differential measurements of ν_d, however we found that the new ν_d values for ^{235}U and ^{238}U were in good agreement with those obtained by the adjustment.

For ^{235}U, we surveyed experimental data of ν_d at the thermal energy, and the experimental data of Borzakov *et al.*[98], Reeder and Warner[99], Synetos and Williams[100], Conant and Palmedo[101], and Keepin *et al.*[102] were used to obtain the averaged value of 0.01585. This is about 1% smaller than the value of 0.01600 in JENDL-3.2, and very close to the value (0.01583) obtained by the adjustment[97]. Note that some old measurements were re-normalized by Tuttle[103].

Above the thermal energy, the whole energy range was split into four energy regions to represent the energy variation of ν_d. The experimental data of Loaiza *et al.*[104], Gudkov *et al.*[105], Besant *et al.*[106], Cox[107], Evans *et al.*[108], Krick and Evans[109], Masters *et al.*[110], Maksyutenko *et al.*[111], Keepin *et al.*[102], Bobkov *et al.*[112], and Keepin[113] were included. The least-squares fitting code GMA[39] was used, and the covariance matrices were also obtained at the same time. The comparison of ν_d of ^{235}U in JENDL-3.2 and JENDL-3.3 is shown in Fig. 26.

For ^{238}U, we found that the experimental data of Krick and Evans[109] were normalized to an older measurement of Masters *et al.*[110]. We renormalized the experimental data of Krick and Evans[109] and Masters *et al.*[110] to a newer value of Meadows[114], and carried out the least-squares fitting to those data with the SOK code[36]. The evaluated ν_d for ^{238}U is shown in Fig. 27. Although measurements of Maksytenko *et al.*[115] are plotted in this figure, those data were not used for evaluation because no uncertainties were given.

The ν_d of ^{233}U was directly obtained by using the least-squares fitting to the experimental data available. The GMA code was used to evaluate ν_d covariances. At the thermal energy the evaluation was based on the experimental data of Conant and Palmedo[101]. Above this energy the experimental data of Evans *et al.*[108], Krick and Evans[109], Keepin *et al.*[102], Rose and Smith[116], Brunson *et al.*[117], Keepin[113], and Masters*et al.*[110] were included. The comparison of ν_d of ^{233}U in JENDL-3.2 and JENDL-3.3 is shown in Fig. 28.

The ν_d value of ^{239}Pu was untouched, since we did not find any reason to change the value for the current evaluation of JENDL-3.2.

Comparisons of the ν_d values for ^{235}U, ^{238}U, and ^{239}Pu at thermal energy with those values in JENDL-3.2 and ENDF/B-VI, as well as the recommendation values by the Delayed Neutron Working Group[97] are shown in Table 10.

The delayed neutron spectra χ_d of Saphier *et al.*[118], which were adopted for ^{232}Th, ^{233}U, ^{235}U, ^{238}U, and ^{239}Pu in JENDL-3.2, were replaced by the calculated values of Brady and England[119] for JENDL-3.3. For other nuclides, their data were also adopted. Therefore the χ_d in JENDL-3.3 are the same as those in ENDF/B-VI.

The temporal six group constants (decay constants and abundances) were determined by the Delayed Neutron Working Group for ^{235}U, ^{238}U and ^{239}Pu[97]. They adopted eight group constants recommended by Spriggs *et al.*[120], and transformed into the six group representation. In this procedure, only one set of the decay constants was used, because of the constraint of ENDF format. For the other nuclides, the decay constants recommended by Brady and England[119] at fast neutron energy were adopted.

Table 10: Number of delayed neutrons per fission at thermal energy. The fourth row shows the recommendation values of Delayed Neutron Working Group.

	^{235}U	^{238}U	^{239}Pu
JENDL-3.2	0.01600	0.04810	0.00622
JENDL-3.3	0.01585	0.04634	0.00622
ENDF/B-VI	0.01670	0.04400	0.00645
Delayed Neutron WG	0.01583	0.04660	0.00639

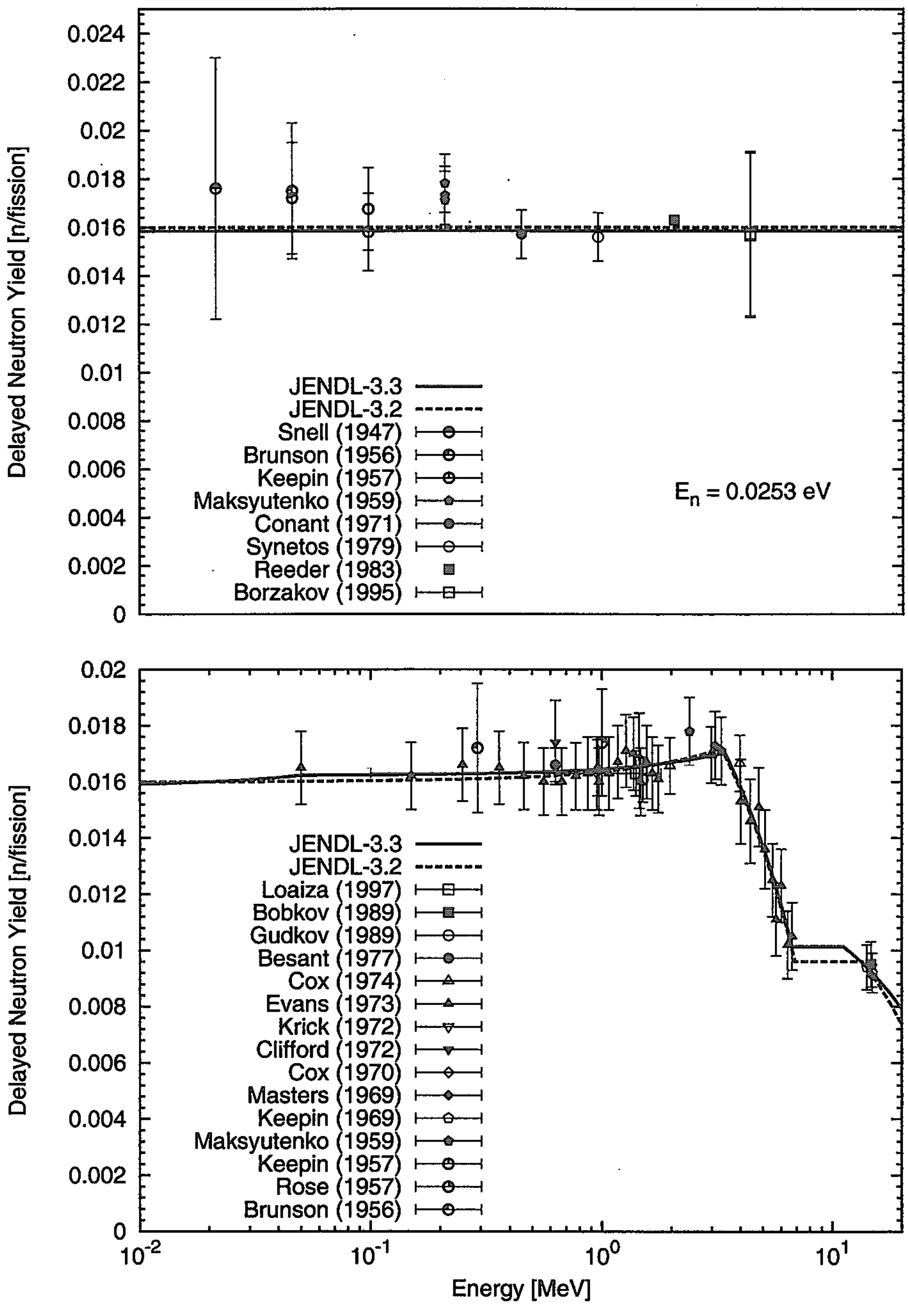

Fig. 26: Number of delayed neutrons per fission ν_d for ^{235}U.

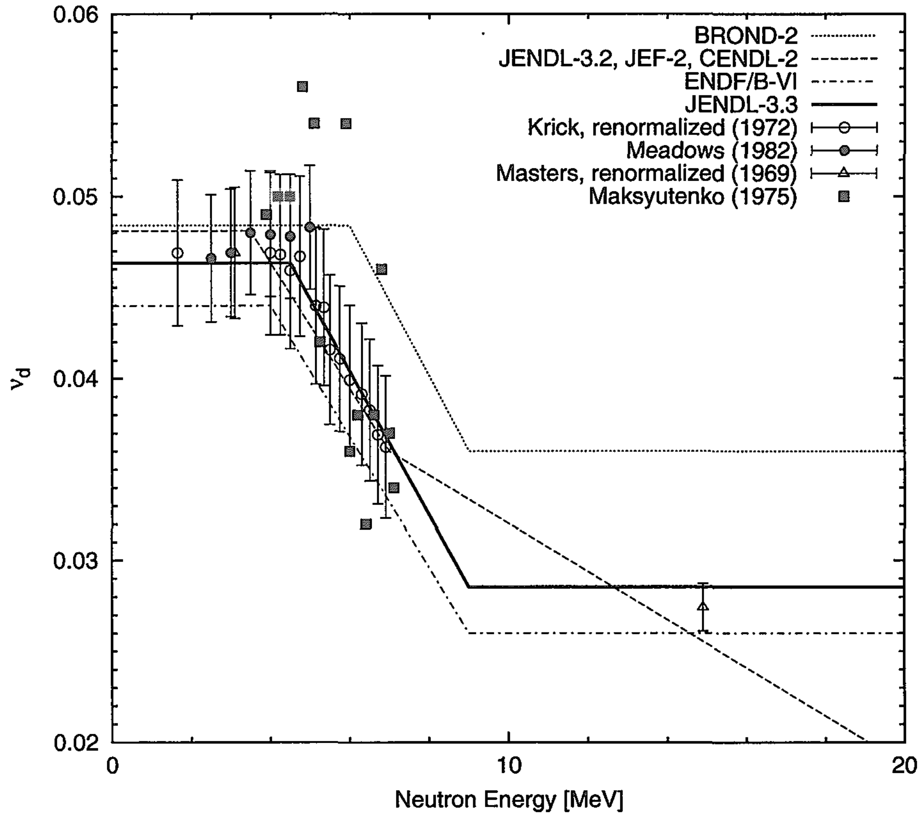

Fig. 27: Number of delayed neutrons per fission ν_d for ^{238}U. The experimental data of Krick and Evans[109] and Masters *et al.*[110] were renormalized to those of Meadows[114].

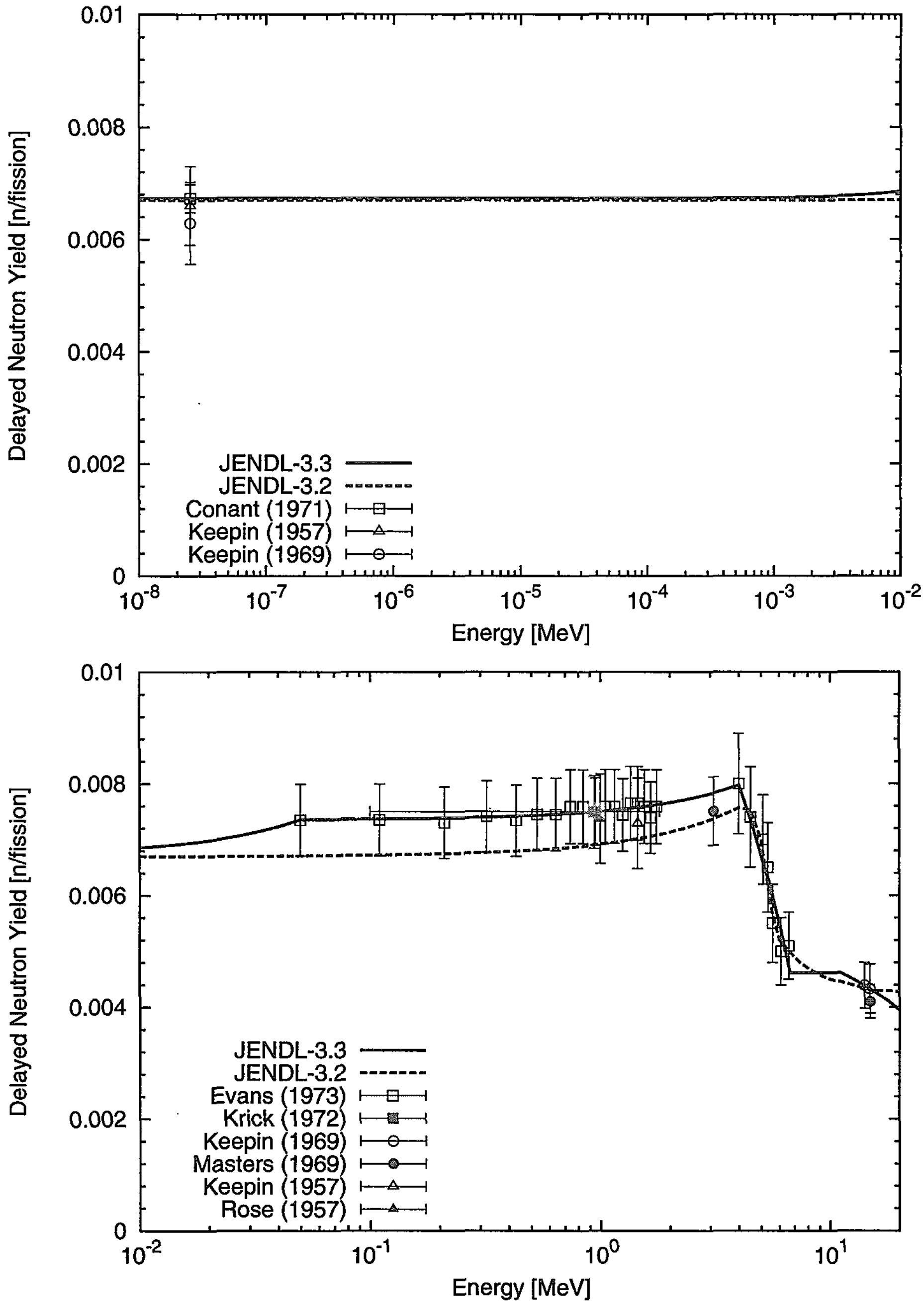

Fig. 28: Number of delayed neutrons per fission ν_d for ^{233}U.

3 COVARIANCE DATA

JENDL-3.3 is featured by providing covariance data for important nuclides such as ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, and ^{241}Pu. Covariance Evaluation Working Group[121] in JNDC has developed techniques and tools for the covariance evaluation, and they have done the covariance evaluations for JENDL-3.2. In some cases we were able to transfer the covariance data from JENDL-3.2 to 3.3 if the data were the same. Otherwise new covariance evaluations were performed for the current release.

In general, the evaluation methods are the same as the procedure adopted in the JENDL-3.2 Covariance File[121]. We applied a generalized least-squares method or a parameter adjustment with the KALMAN system[37] to obtain the covariances for various quantities.

The covariances of resolved resonance parameters are given for ^{238}U and ^{239}Pu. Those were evaluated with the simplified method developed by Kawano and Shibata[11], and stored in the JENDL-3.2 covariance file. In the case of ^{235}U, technically the same method can be applied to the resolved resonance parameters of Leal *et al.*[15]. However it is very difficult to perform this because there are a large number of parameters. This problem is now under discussion in the framework of the OECD/NEA Working Party on International Evaluation Cooperation (WPEC).

The simultaneous evaluation of fission cross sections[8, 36] enabled us to prepare uncertainties in the cross sections, correlations between different energy points, and correlations among different reactions such as $^{235}\mathrm{U}(n,f)$ *vs.* $^{239}\mathrm{Pu}(n,f)$.

4 Conclusion

We started Heavy Nuclide Data Evaluation Working Group under Japanese Nuclear Data Committee in 1998, and the latest JENDL — JENDL-3.3 — was released in May 2002. In this report we described major changes in the heavy nuclide data we had performed during this period, though there are some minor revisions which were not mentioned here. Apparently the revisions were rather extensive.

A validation test of JENDL-3.3 was carried out by Takano, Nakagawa, and Kaneko[38] with the continuous energy Monte Carlo method, and they reported that JENDL-3.3 settled the k_{eff} overestimation problem. We believe that the heavy nuclide data in JENDL-3.3 are satisfactory for various nuclear applications.

Finally it should be noted that we have got many feedbacks from integral tests during our evaluations such as the benchmark calculations for thermal and fast reactors, β_{eff} measurements, fission-spectrum-averaged cross sections, *etc.* Although the evaluation is completely based on the differential data, such collaborations were a great help to establish the reliability of JENDL-3.3 heavy nuclide data.

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国際単位系（SI）と換算表

表1　SI基本単位および補助単位

量	名　称	記　号
長　さ	メートル	m
質　量	キログラム	kg
時　間	秒	s
電　流	アンペア	A
熱力学温度	ケルビン	K
物質量	モル	mol
光　度	カンデラ	cd
平面角	ラジアン	rad
立体角	ステラジアン	sr

表2　SIと併用される単位

名　称	記　号
分，時，日	min, h, d
度，分，秒	°, ′, ″
リットル	l, L
トン	t
電子ボルト	eV
原子質量単位	u

$1\ \mathrm{eV} = 1.60218 \times 10^{-19}\ \mathrm{J}$

$1\ \mathrm{u} = 1.66054 \times 10^{-27}\ \mathrm{kg}$

表3　固有の名称をもつSI組立単位

量	名　称	記号	他のSI単位による表現
周波数	ヘルツ	Hz	s^{-1}
力	ニュートン	N	$m \cdot kg/s^2$
圧力，応力	パスカル	Pa	N/m^2
エネルギー，仕事，熱量	ジュール	J	N·m
工率，放射束	ワット	W	J/s
電気量，電荷	クーロン	C	A·s
電位，電圧，起電力	ボルト	V	W/A
静電容量	ファラド	F	C/V
電気抵抗	オーム	Ω	V/A
コンダクタンス	ジーメンス	S	A/V
磁束	ウェーバ	Wb	V·s
磁束密度	テスラ	T	Wb/m^2
インダクタンス	ヘンリー	H	Wb/A
セルシウス温度	セルシウス度	℃	
光束	ルーメン	lm	cd·sr
照度	ルクス	lx	lm/m^2
放射能	ベクレル	Bq	s^{-1}
吸収線量	グレイ	Gy	J/kg
線量当量	シーベルト	Sv	J/kg

表4　SIと共に暫定的に維持される単位

名　称	記　号
オングストローム	Å
バーン	b
バール	bar
ガル	Gal
キュリー	Ci
レントゲン	R
ラド	rad
レム	rem

$1\ \mathrm{Å} = 0.1\ \mathrm{nm} = 10^{-10}\ \mathrm{m}$

$1\ \mathrm{b} = 100\ \mathrm{fm}^2 = 10^{-28}\ \mathrm{m}^2$

$1\ \mathrm{bar} = 0.1\ \mathrm{MPa} = 10^5\ \mathrm{Pa}$

$1\ \mathrm{Gal} = 1\ \mathrm{cm/s^2} = 10^{-2}\ \mathrm{m/s^2}$

$1\ \mathrm{Ci} = 3.7 \times 10^{10}\ \mathrm{Bq}$

$1\ \mathrm{R} = 2.58 \times 10^{-4}\ \mathrm{C/kg}$

$1\ \mathrm{rad} = 1\ \mathrm{cGy} = 10^{-2}\ \mathrm{Gy}$

$1\ \mathrm{rem} = 1\ \mathrm{cSv} = 10^{-2}\ \mathrm{Sv}$

表5　SI接頭語

倍数	接頭語	記　号
10^{18}	エクサ	E
10^{15}	ペタ	P
10^{12}	テラ	T
10^{9}	ギガ	G
10^{6}	メガ	M
10^{3}	キロ	k
10^{2}	ヘクト	h
10^{1}	デカ	da
10^{-1}	デシ	d
10^{-2}	センチ	c
10^{-3}	ミリ	m
10^{-6}	マイクロ	μ
10^{-9}	ナノ	n
10^{-12}	ピコ	p
10^{-15}	フェムト	f
10^{-18}	アト	a

（注）

1. 表1－5は「国際単位系」第5版，国際度量衡局 1985年刊行による。ただし，1 eV および1 uの値はCODATAの1986年推奨値によった。
2. 表4には海里，ノット，アール，ヘクタールも含まれているが日常の単位なのでここでは省略した。
3. barは，JISでは流体の圧力を表わす場合に限り表2のカテゴリーに分類されている。
4. EC閣僚理事会指令ではbar，barnおよび「血圧の単位」mmHgを表2のカテゴリーに入れている。

換　算　表

力	N（$=10^5$dyn）	kgf	lbf
	1	0.101972	0.224809
	9.80665	1	2.20462
	4.44822	0.453592	1

粘　度　1 Pa·s（$N \cdot s/m^2$）= 10 P（ポアズ）（g/(cm·s)）

動粘度　$1\ m^2/s = 10^4$St（ストークス）（cm^2/s）

圧力	MPa（=10 bar）	kgf/cm²	atm	mmHg（Torr）	lbf/in²（psi）
	1	10.1972	9.86923	7.50062×10^3	145.038
	0.0980665	1	0.967841	735.559	14.2233
	0.101325	1.03323	1	760	14.6959
	1.33322×10^{-4}	1.35951×10^{-3}	1.31579×10^{-3}	1	1.93368×10^{-2}
	6.89476×10^{-3}	7.03070×10^{-2}	6.80460×10^{-2}	51.7149	1

エネルギー・仕事・熱量	J（$=10^7$ erg）	kgf·m	kW·h	cal（計量法）	Btu	ft·lbf	eV
	1	0.101972	2.77778×10^{-7}	0.238889	9.47813×10^{-4}	0.737562	6.24150×10^{18}
	9.80665	1	2.72407×10^{-6}	2.34270	9.29487×10^{-3}	7.23301	6.12082×10^{19}
	3.6×10^6	3.67098×10^5	1	8.59999×10^5	3412.13	2.65522×10^6	2.24694×10^{25}
	4.18605	0.426858	1.16279×10^{-6}	1	3.96759×10^{-3}	3.08747	2.61272×10^{19}
	1055.06	107.586	2.93072×10^{-4}	252.042	1	778.172	6.58515×10^{21}
	1.35582	0.138255	3.76616×10^{-7}	0.323890	1.28506×10^{-3}	1	8.46233×10^{18}
	1.60218×10^{-19}	1.63377×10^{-20}	4.45050×10^{-26}	3.82743×10^{-20}	1.51857×10^{-22}	1.18171×10^{-19}	1

1 cal = 4.18605 J（計量法）
　= 4.184 J　（熱化学）
　= 4.1855 J（15 °C）
　= 4.1868 J（国際蒸気表）

仕事率　1 PS（仏馬力）
　= 75 kgf·m/s
　= 735.499 W

放射能	Bq	Ci
	1	2.70270×10^{-11}
	3.7×10^{10}	1

吸収線量	Gy	rad
	1	100
	0.01	1

照射線量	C/kg	R
	1	3876
	2.58×10^{-4}	1

線量当量	Sv	rem
	1	100
	0.01	1

（86年12月26日現在）

Evaluations of Heavy Nuclide Data for JENDL-3.3