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Toshihiko KAWANO*, Allan D. CARLSON*, Hiroyuki MATSUNOBU*,
Tsuneo NAKAGAWA, Keiichi SHIBATA, Patrick TALOU*,
Philip G. YOUNG* and Mark B. CHADWICK*

日本原子力研究所
Japan Atomic Energy Research Institute

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Tsuneo NAKAGAWA, Keiichi SHIBATA,
Patrick TALOU[*4], Philip G. YOUNG[*4] and Mark B. CHADWICK[*4]

Department of Nuclear Energy System
Tokai Research Establishment
Japan Atomic Energy Research Institute
Tokai-mura, Naka-gun, Ibaraki-ken

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Comparisons of evaluated fission cross sections for ^{235}U in JENDL-3.3 and ENDF/B-VI are carried out. The comparisons are made for both the differential and integral data. The fission cross sections as well as the fission ratios are compared with the experimental data in detail. Spectrum averaged cross sections are calculated and compared with the measurements. The employed spectra are the ^{235}U prompt fission neutron spectrum, the ^{252}Cf spontaneous fission neutron spectrum, and the neutron spectrum produced by a ^{9}Be(d,xn) reaction. For ^{235}U prompt fission neutron spectrum, the ENDF/B-VI evaluation reproduces experimental averaged cross sections. For ^{252}Cf and ^{9}Be(d,xn) neutron spectra, the JENDL-3.3 evaluation gives better results than ENDF/B-VI.

Keywords: Fission Cross Section, Standard Cross Section, Spectrum Averaged Cross Section, Uranium 233, Uranium 235, Uranium 238, Plutonium 239, Plutonium 240, Plutonium 241

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*1 Kyushu University
*2 National Institute of Standards and Technology
*3 Data Engineering, Inc.
*4 Los Alamos National Laboratory

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JENDL-3.3 と ENDF/B-VI の ^{235}U 核分裂断面積の比較

日本原子力研究所東海研究所エネルギーシステム研究部
河野　俊彦[*1]・Allan D. CARLSON[*2]・松延　廣幸[*3]
中川　庸雄・柴田　恵一
Patrick TALOU[*4]・Philip G. YOUNG[*4]・Mark B. CHADWICK[*4]

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JENDL-3.3 と ENDF/B-VI の ^{235}U 核分裂断面積の比較を、微分データと積分データの両方に対して行う。核分裂断面積と核分裂比のデータを、実験データと詳細に比較する。また、スペクトル平均断面積を計算し、実験値と比較する。用いたスペクトルは、^{235}U 即発核分裂中性子スペクトル、^{252}Cf 自発核分裂中性子スペクトル、$^{9}Be(d,xn)$反応中性子スペクトルである。^{235}U 核分裂スペクトルによる平均断面積では、ENDF/B-VI の評価値が実験データを再現する。一方、^{252}Cf 自発核分裂と $^{9}Be(d,xn)$反応の中性子スペクトルでは、JENDL-3.3 の方が ENDF/B-VI よりも良好な結果を与える。

東海研究所：〒319-1195 茨城県那珂郡東海村白方白根 2-4
*1 九州大学
*2 国立標準技術研究所
*3 （株）データ工学
*4 ロスアラモス国立研究所

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1 INTRODUCTION

Since fission cross sections of ^{235}U are probably the most important quantities for applications of nuclear technology, special attention must be paid for their evaluation. At the thermal energy or in the epithermal energy region, those cross sections are relatively well-determined because precise resonance analyses are available[1, 2]. In the unresolved energy region and the smooth region, each resonance cannot be resolved, and the resonance analysis is no longer available. Therefore a least-squares fitting to experimental data with simple functions has been adopted for the evaluations of fission cross sections. With this technique the evaluated cross section can be obtained without any ambiguities because this is a mere "averaged value" of the experimental data.

A simultaneous evaluation of fission cross sections of Uranium and Plutonium isotopes was carried out for JENDL-3.3[3, 4], and the fission cross sections for ^{235}U were determined in the energy range 30 keV – 20 MeV. A similar simultaneous evaluation[5] combined with R-matrix evaluations was performed for ENDF/B-VI, and the fission cross sections of ^{235}U obtained were regarded as a "standard cross section" in the ENDF project. 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. 1. In this figure the cross sections are represented by a 640-group structure.

The difference in the fission cross sections in the MeV energy region is less important for development of thermal reactors, but the cross sections there are essential in a more general sense because most of the neutrons produced by a fission are in the MeV energy region. In addition, inaccurate ^{235}U fission cross sections may distort other cross sections when one uses them as a standard.

In order to investigate the difference in the ^{235}U fission cross sections between JENDL-3.3 and ENDF/B-VI, we surveyed this issue from various aspects — a detailed comparison with the experimental data, comparisons with integral quantities, etc. In this report we do not intend to give the best values for the ^{235}U fission cross sections, but to provide information on the accurate evaluation for the standards.

2 Comparison with Differential Data

2.1 Magnified Plot near 2 MeV

Detailed comparisons of evaluated cross sections in the energy range of 1–4 MeV with the experimental data are shown in Figs. 2–15. The comparison involves cross sections of ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, and ^{241}Pu. In Figs. 2–7 absolute cross sections are plotted, and the ratio measurements are in Figs. 8–15. The solid lines are the cross sections in JENDL-3.3, and the dotted lines are those in ENDF/B-VI. The experimental data which have been plotted are all shown by the same symbol (•) without error-bars.

It is difficult to conclude which evaluation is better from these drawings. Some figures show that ENDF/B-VI gives a better fit to the data, and vice versa. An overestimation of ENDF/B-VI is seen for the fission ratio of ^{240}Pu to ^{235}U in Fig. 12 (Fig. 13 for close-up), although the absolute fission cross sections of ^{240}Pu in the two evaluations are very close as is seen in Fig. 6— the difference is less than 2% in this energy range[4]. Since the JENDL-3.3 evaluation included ^{240}Pu data but the ENDF/B-VI standards evaluation did not, the consistency of JENDL-3.3 for ^{240}Pu data are natural. It can be noted that the consistency between ^{235}U and ^{240}Pu data in ENDF/B-VI should be improved.

2.2 Energy Grid Setting

In Fig. 3 a systematic difference can be seen in the energy range 1–4 MeV. This difference continues up to 6 MeV — ENDF/B-VI is lower than JENDL-3.3 about 2–3%, and at 1.2 MeV it is about 3.2%[3, 4]. One of the reasons for this large difference at 1.2 MeV is energy grid used for the evaluation. In the JENDL-3.3 evaluation the energy grid includes points at 1.0 MeV and 1.5 MeV, and linear interpolation was used between those two energy points. After that, new energy points were re-defined by means of cubic-spline interpolation in order to obtain a smooth curve. The spline interpolation gives a convex curve near 1.2 MeV, and this results in somewhat higher cross sections between these two energy points. If the energy grids are linear-interpolated the difference becomes smaller (2.8%) but still remains.

It is interesting to see what happens if one adds a new energy point between 1.0 and 1.5 MeV. We repeated the simultaneous evaluation for JENDL-3.3, but a new energy grid was set including 1.25 MeV. The ^{235}U cross sections obtained are shown in Table 1. The cross section for JENDL was lowered by 24 mb (1.9%) at 1.25 MeV and became closer to ENDF, while at 1.0 and 1.5 MeV the differences became larger. It can be noted that the higher cross sections in JENDL-3.3 due to the cubic-spline interpolation near 1.2 MeV should be corrected.

2.3 Comparisons with Recent Measurements

To make comparisons with the experimental data in detail, some selected data (relatively new ones) are plotted. The experimental data are Johnson et al.[6], Lisowski et al.[7], Kalinin et al.[8], and Staples et al.[9]

Comparisons of ^{235}U fission cross sections with the data of Johnson et al.[6], Lisowski et al.[7], and Kalinin et al.[8] are shown in Figs. 16 and 17. These two plots are the same but have different scales. One can clearly see in Fig. 17 that ENDF/B-VI gives an excellent fit to Johnson's data from 1 to 2 MeV, while ENDF seems to underestimate above 2 MeV. JENDL gives opposite tendency. In the energy range 1–2 MeV the cross sections in JENDL-3.3 overestimate the measurements, but JENDL-3.3 gives a better fit than ENDF/B-VI above 2 MeV.

Comparisons with the Staples' data are shown in Figs. 18–21. Figures 18, 19 are for the fission ratio of ^{239}Pu to ^{235}U, and Figs. 20, 21 are for ^{240}Pu, respectively. A tendency of overestimation can be seen for ENDF in Figs. 19 and 21.

3 Comparison with Integral Data

3.1 ^{235}U Prompt Fission Neutron Spectrum Averaged Cross Section

An averaged cross section in a well-determined neutron field provides a good integral test of cross sections. The ^{235}U prompt fission neutron spectrum at thermal energy is often used for this purpose. This integral test is suitable for threshold reactions since contributions from low energy neutrons are cut off. In the case of the ^{235}U fission cross section, neutrons for the entire energy range contribute to the averaged cross sections. Contributions from the resolved/unresolved resonance regions should be examined first.

We calculate a fractional contribution of the $\sigma_f \chi$ values to the averaged cross section. This is expressed as

$$f(E) = \frac{1}{\overline{\sigma}_f} \int_0^E \sigma_f(\epsilon)\chi(\epsilon) d\epsilon, \tag{1}$$

where $\chi(\epsilon)$ is the fission neutron spectrum. The fission cross sections in JENDL-3.3 and ENDF/B-VI are expressed in the 640-group structure, as shown in Fig. 1. The calculated fractional contribution $f(E)$ is shown in Fig. 22 as a function of upper-limit integration energy E. The upper border of the unresolved resonance region in JENDL-3.3 is 30 keV, and from Fig. 22 the fractional contribution below 30 keV is less than 1%. Therefore the influence of the resonance region is negligible.

A calculation of the averaged cross section also depends on the fission spectrum used. The prompt fission neutron spectrum was re-evaluated for JENDL-3.3 by Ohsawa et al.[10, 11] with multi-modal fission analysis. We employ the spectrum by Ohsawa et al. for the calculation of averaged cross sections. The difference between the Ohsawa et al. spectrum and that in ENDF/B-VI at thermal energy is very small. The neutron spectra in the two evaluations are compared in Fig. 23. The solid line is the evaluated spectrum in JENDL-3.3, and the dashed line is that of ENDF/B-VI.

A quantity $\chi\sigma$ which represents a fractional contribution to the averaged cross section is plotted as a function of energy in Fig. 24. As can be seen from this figure, the fission cross sections near 1.5 MeV dominate the averaged values.

The calculated fission spectrum averaged cross sections are compared with the evaluated value by Mannhart[12] in Table 2. The ratios of the evaluations to the evaluated experimental data by Mannhart are, 0.997 for ENDF/B-VI and 1.015 for JENDL-3.3.

3.2 ^{252}Cf Spontaneous Fission Neutron Spectrum Averaged Cross Section

The ^{252}Cf spontaneous fission spectrum is often used for integral tests of cross sections since the spectrum is well-determined[13]. In Fig. 23 the evaluated ^{252}Cf spectrum is compared with the ^{235}U prompt fission neutron spectra. The ^{252}Cf spectrum is harder than the ^{235}U spectrum.

There exist several experimental data for the $^{235}\mathrm{U}(n,f)$ cross section averaged over the ^{252}Cf spectrum[14, 15, 16, 17, 18], and their uncertainties are relatively small. The experimental data are mainly from the National Bureau of Standards (NBS), and are listed in Table 3. Those integral data were included in the ENDF/B-VI evaluation[5], except for Grundl et al.[14]

Calculated averaged cross sections are, 1.217 b for ENDF/B-VI and 1.238 b for JENDL-3.3. Comparison of the calculated cross sections with the experimental data is shown in Fig. 25. The ENDF/B-VI value is close to the measured value of Heaton et al.[15] and Davis et al.[16], while the value for JENDL-3.3 is consistent with the data of Schroder et al.[17]. The newest data among them is that of Schroder et al.[17], which is higher than that of Grundl et al. and Heaton et al. although those measurements were carried out at the same institute. However the difference is still within the size of the error-bars.

The mean value of the experimental data is 1.233±0.009 b, and the C/E values for this value are, 0.987 for ENDF/B-VI and 1.004 for JENDL-3.3.

3.3 $^{9}\mathrm{Be}(d,xn)$ Neutron Spectrum Averaged Cross Section

Averaged cross sections in a $^{9}\mathrm{Be}(d,xn)$ neutron field were measured by Watanabe et al.[19] at ANL. This neutron spectrum has an interesting feature — most of the neutrons have energies between 1 and 6 MeV. So that the averaged cross sections give us some criteria for the fission cross sections in the energy range of interest. The neutron spectrum[20] is shown in Fig. 26.

Since the calculated averaged cross section depends both on the cross section and the spectrum, uncertainties in the spectrum cause some errors in the calculation. The authors in Ref. [19] investigated this by testing several neutron spectra which were measured by an independent method, and concluded that the averaged values may have uncertainties of $< 1\%$.

With the neutron spectrum reported by the ANL group[19, 20], we calculated the averaged fission cross sections for JENDL-3.3 and ENDF/B-VI. Calculated values are, 1.203 b for ENDF/B-VI and 1.225 b for JENDL-3.3. The experimental data of Watanabe et al. is 1.217 b. The C/E values are 0.988 and 1.007 for ENDF/B-VI and JENDL-3.3, respectively.

3.4 GODIVA Benchmark Test

A criticality benchmark test was carried out by Takano et al.[21] at JAERI. The continuous energy Monte Carlo code MVP was used. To investigate the fission cross sections of ^{235}U in the fast neutron energy region, calculations for a small-size fast neutron core are examined. The fast neutron core GODIVA is a bare sphere of highly enriched U, therefore differences in the fission cross sections directly affect the k_{eff} value. The C/E value for JENDL-3.3 is 1.0032, while ENDF/B-VI is 0.9965. Note that this value depends not only on the fission cross sections but also on the other nuclear data such as prompt neutron fission spectra, inelastic scattering cross sections, etc. The fission spectra for JENDL-3.3 and ENDF/B-VI are very similar, but the inelastic scattering cross sections are different.

4 Results and Discussions

We have shown the difference in the ^{235}U fission cross sections in the JENDL-3.3 and ENDF/B-VI evaluations from two aspects, the differential and integral data. Spectrum averaged cross sections in several neutron fields were calculated and compared with the experimental data. The C/E values for those integral data are summarized in Table 4. Results of a criticality benchmark test[21] for GODIVA are also shown in Table 4.

A main reason for the difference in the two evaluations — JENDL-3.3 and ENDF/B-VI — is the experimental databases used in the evaluations. For example any integral-type data were not included in the JENDL-3.3 evaluation, while ^{252}Cf spectrum averaged data for the $^{235}\mathrm{U}(n,f)$ and $^{239}\mathrm{Pu}(n,f)$ cross sections were included in the ENDF/B-VI standards evaluation. These data may act as a normalization for the evaluations.

There are some other quantities those were included in the ENDF/B-VI evaluation but not used in the JENDL-3.3 evaluation. Those are cross section data (and ratios) for $^{238}\mathrm{U}(n,\gamma)$, $^{197}\mathrm{Au}(n,\gamma)$, $^{10}\mathrm{B}(n,\alpha)$, $^{10}\mathrm{B}(n,\alpha_1\gamma)$, $^{10}\mathrm{B}(n,n)$, $^{10}\mathrm{B}(n,\mathrm{total})$, $^{6}\mathrm{Li}(n,t)$, $^{6}\mathrm{Li}(n,n)$, $^{6}\mathrm{Li}(n,\mathrm{total})$ and the thermal constants. The impact of those quantities on the calculated results is not clear at this moment, but we do not expect that the ratios to the ^{197}Au, ^{10}B, and ^{6}Li cross sections will have a big impact on the calculated results because the number of data points are less than those in the fission cross section measurements. Anyway the effect of those data should be examined in detail.

In the JENDL-3.3 evaluation, cross sections for $^{233}\mathrm{U}(n,f)$, $^{240}\mathrm{Pu}(n,f)$ and $^{241}\mathrm{Pu}(n,f)$ were included, but those were not used in the ENDF/B-VI evaluation. In addition some new measurements were included in the JENDL-3.3 evaluation, namely, $^{233}\mathrm{U}/^{235}\mathrm{U}$ fission ratio data by Shcherbakov[22], $^{239}\mathrm{Pu}/^{235}\mathrm{U}$ and $^{240}\mathrm{Pu}/^{235}\mathrm{U}$ fission ratio data by Staples and Morley[9]. For older measurements, some cross section data were excluded in the JENDL-3.3 evaluation because they are inconsistent. While in the ENDF/B-VI evaluation, the effect of such inconsistent data was handled in GMA by down-weighting them if they were more than 3 standard deviations away from the output results. These different treatments for such data could produce some differences in the two evaluations.

In the JENDL-3.3 evaluation the thermal constants would not have an effect and was not included. On the other hand, the ENDF/B-VI evaluation made use of data sets which extended to thermal energies. Also there were cases where a number of sets would overlap and their full range extended from very high energies to thermal energies. Thus the thermal constant data had an impact on that evaluation in that they helped to define the normalization of high energy data (as well as low energy data). Since the JENDL-3.3 evaluation did not use the thermal constant data, in principle, this could lead to differences in the two evaluations.

It is probably worth recalling the different treatment of experimental data in the JENDL-3.3 and ENDF/B-VI evaluations. In the simultaneous evaluation which was carried out by the JENDL project[3, 4], all the measurements were categorized into two groups — the absolute measurements and the relative measurements. The uncertainties in the experimental data were carefully re-evaluated in order to reduce discrepancies in experiments. For the cross sections, the reported values were used without any normalizations. In the ENDF/B-VI simultaneous evaluation[5], some data sets were treated as a "shape measurement," for which a normalization constant c was included as a fitting parameter. Therefore, even though the two evaluations employ the same experimental database, the mean values obtained could be different.

The impact of the inclusion of ^{252}Cf spectrum averaged cross section can be checked in a simple way. The fission cross sections in JENDL-3.3 were updated so as to reproduce the ^{252}Cf data by means of a data adjustment technique based on the Bayes' theorem. The results indicated that changes in the fission cross sections were very small. This can be interpreted as follows: the uncertainties of the evaluated fission cross sections are small, and the number of integral measurements is less than the differential data. So that the relative weight of the differential data is larger than that of the integral data.

One can also see the impact of ^{252}Cf data for the simultaneous evaluation in Fig. 25. Though the ENDF/B-VI evaluation uses most of those integral data, its calculated averaged-cross section is lower than the mean-value of the measurements which is shown by the dot-dashed line. Thus the differential data play an important role in the simultaneous evaluation.

We calculated the C/E values for the ^{252}Cf averaged cross section, however, this value strongly depends on the choice of measurements. For example, if one ignores the measurements of Adamov et al.[18] and Schroder et al.[17], the mean value of the experimental data becomes very close to the calculated value when the ENDF/B-VI evaluation is used. Indeed such a choice is artificial, but one must realize that it is quite dangerous to judge from only one integral test. Nevertheless, a comparison of the ^{252}Cf spectrum averaged cross sections may provide a good criterion for our study. A new measurement with modern experimental technique should be done to investigate the ^{235}U fission cross sections.

5 Conclusion

The ^{235}U fission cross sections in the JENDL-3.3 and ENDF/B-VI evaluations and their databases were investigated in order to help to reconcile the difference between them. Various comparisons of the cross sections yielded some features of those evaluations near 2 MeV, though we were not able to judge definitely which one is superior.

Comparisons were made for the differential and integral data. The evaluated fission cross sections as well as the fission ratios of uranium and plutonium isotopes were compared with the experimental data in the energy range 1–4 MeV. Since the experimental data are scattered there, both the evaluations reproduce a trend of the measured values. We found that the JENDL-3.3 evaluation had an "overshoot" problem because of the use of the cubic-spline interpolation. We also noted that there was an inconsistency between ^{240}Pu fission ratio to ^{235}U data in ENDF/B-VI.

Spectrum averaged cross sections 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.

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Table 1: The ^{235}U fission cross sections at 1.0, 1.25, and 1.5 MeV in ENDF/B-VI (the second column) and JENDL-3.3 (the third column). The fourth column shows the resulting cross sections when a new energy grid point at 1.25 MeV is added in the simultaneous evaluation of JENDL-3.3.

	ENDF/B-VI	JENDL-3.3	new point added
σ_f at 1.00 MeV [b]	1.1969	1.2155	1.2272
difference to ENDF/B-VI	—	1.6%	2.5%
σ_f at 1.25 MeV [b]	1.2020	1.2423	1.2184
difference to ENDF/B-VI	—	3.2%	1.3%
σ_f at 1.50 MeV [b]	1.2321	1.2521	1.2606
difference to ENDF/B-VI	—	1.6%	2.3%

Table 2: Comparison of the calculated ^{235}U fission spectrum averaged ^{235}U(n,f) cross section with the compiled value by Mannhart[12]. The prompt fission neutron spectrum at thermal energy was taken from JENDL-3.3.

	$\overline{\sigma}_f$ [b]
ENDF/B-VI	1.216
JENDL-3.3	1.237
Mannhart (1999)	1.219 ± 0.014

Table 3: Comparison of the experimental data for the ^{252}Cf spontaneous fission spectrum averaged ^{235}U(n,f) cross section.

EXFOR	First Author	Institute Code	Year	$\overline{\sigma}_f$ [b]	Ref.
10304 002	Grundl	USANBS	1972	1.207 ± 0.052	[14]
10809 002	Heaton	USANBS	1976	1.216 ± 0.019	[15]
10698 002	Davis	USAMHG	1978	1.215 ± 0.022	[16]
12953 002	Schroder	USANBS	1985	1.234 ± 0.017	[17]
40547 004	Adamov	CCPRI	1977	1.266 ± 0.019	[18]

Table 4: C/E values for the spectrum averaged fission cross sections. The fifth row shows results of a criticality benchmark test 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

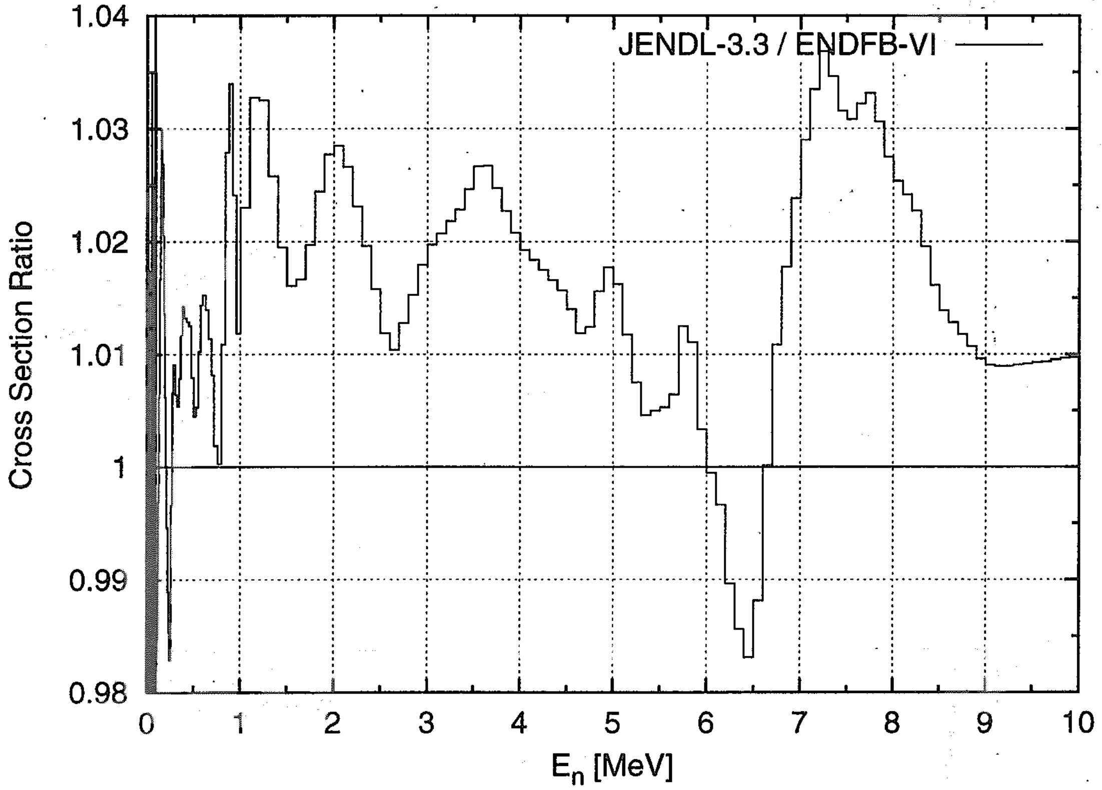

Fig. 1: Ratio of ^{235}U fission cross sections in JENDL-3.3 to those in ENDF/B-VI.

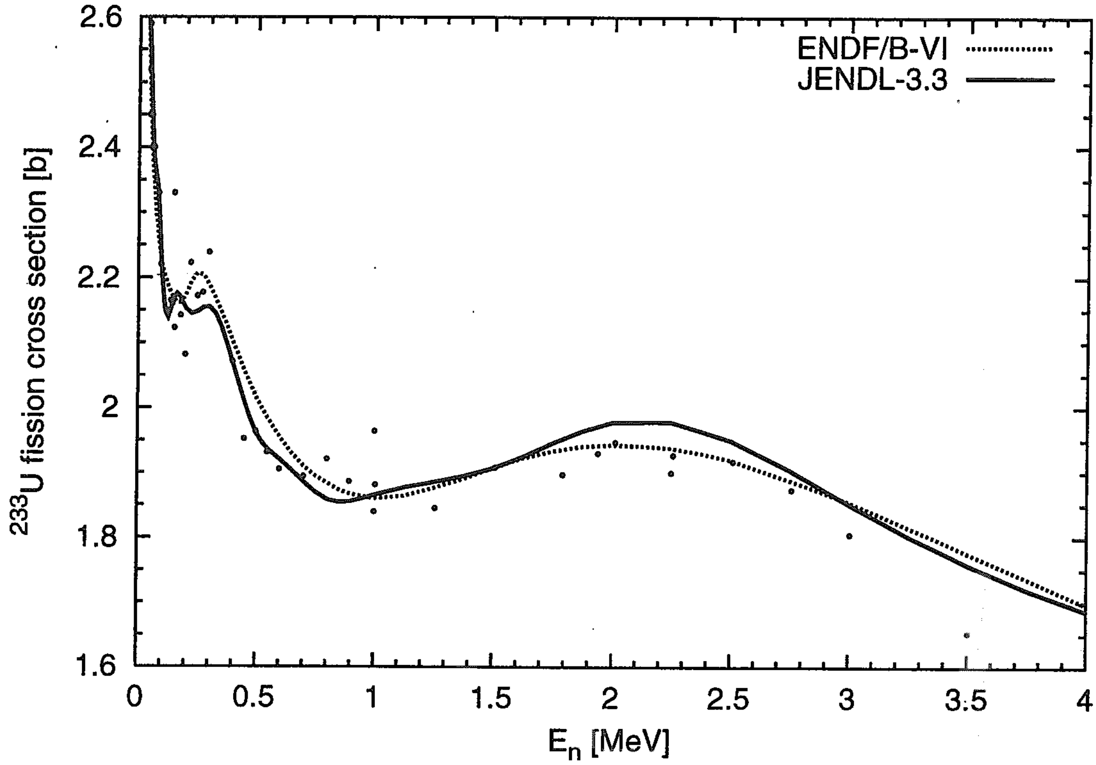

Fig. 2: ^{233}U fission cross sections.

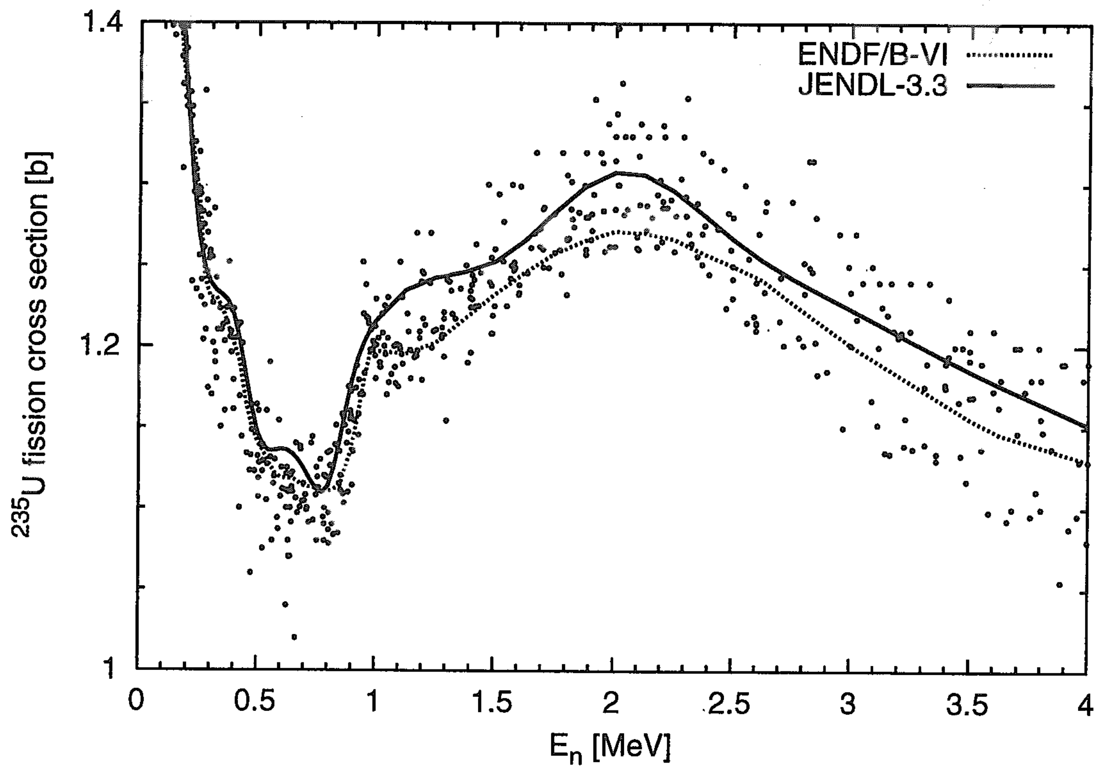

Fig. 3: ^{235}U fission cross sections.

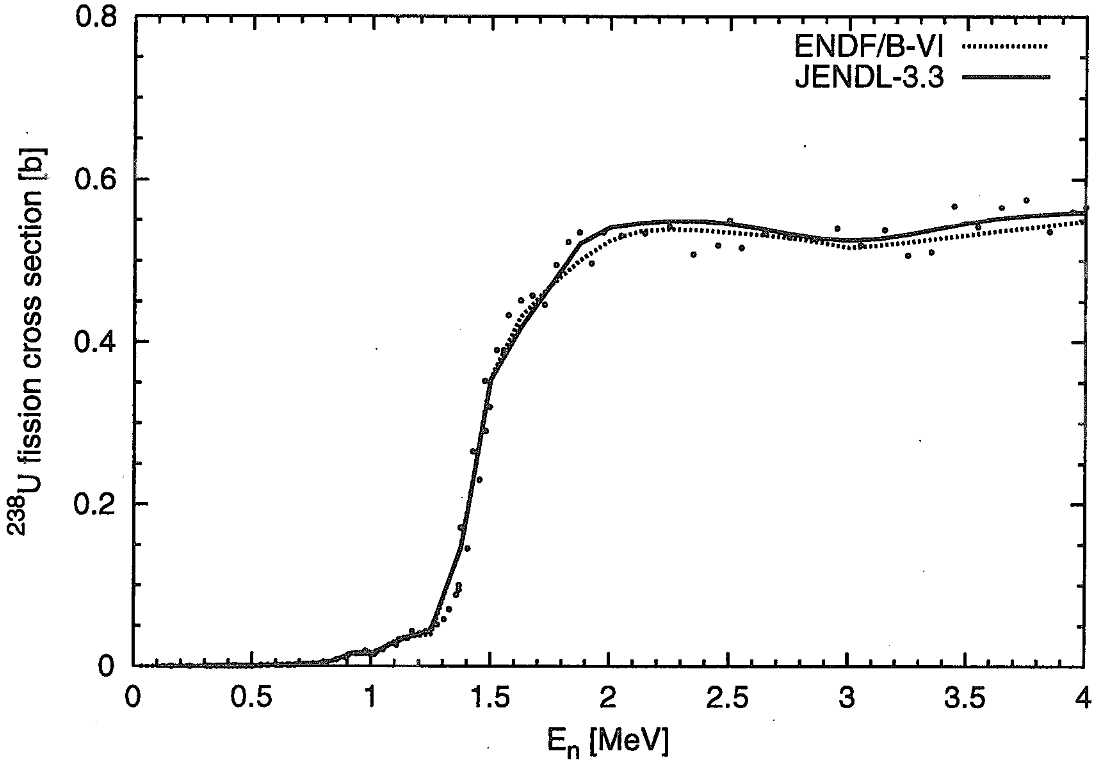

Fig. 4: ^{238}U fission cross sections.

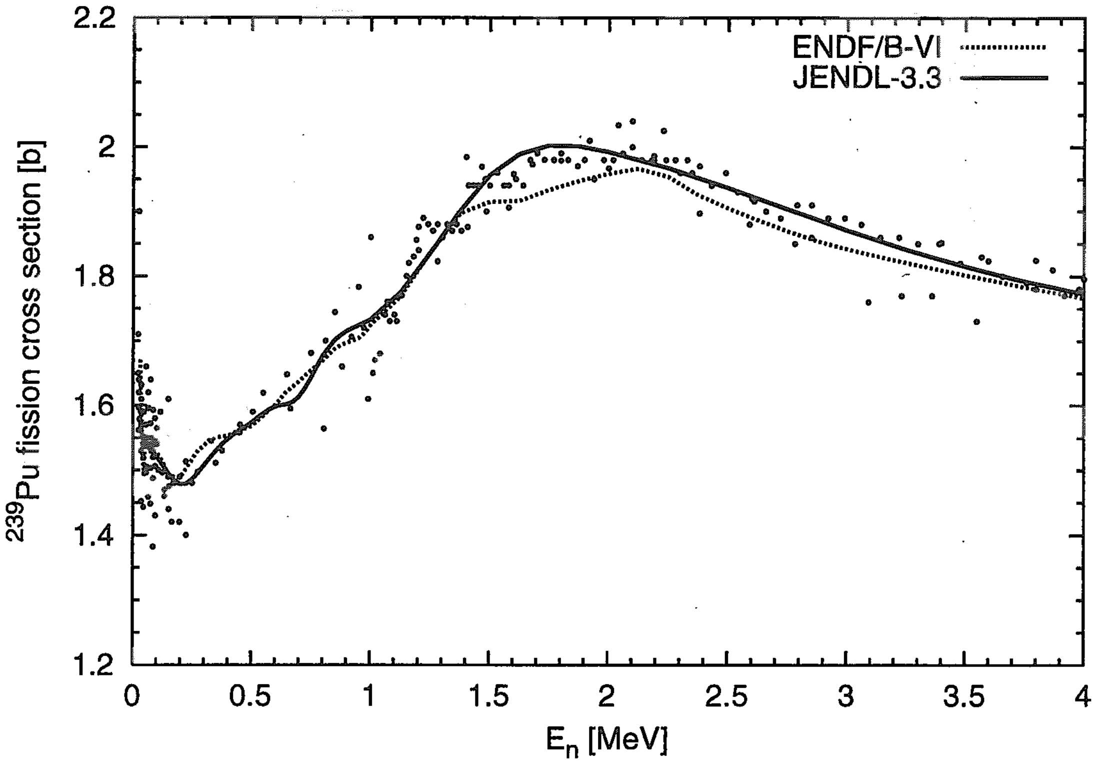

Fig. 5: ^{239}Pu fission cross sections.

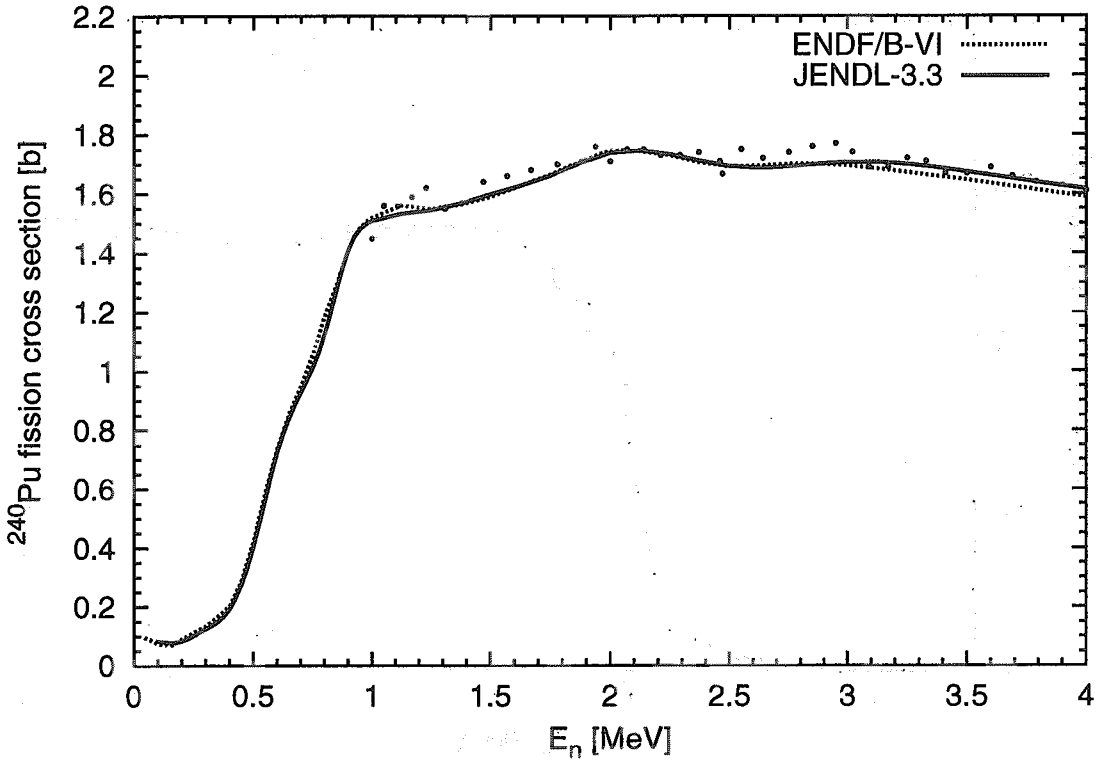

Fig. 6: ^{240}Pu fission cross sections.

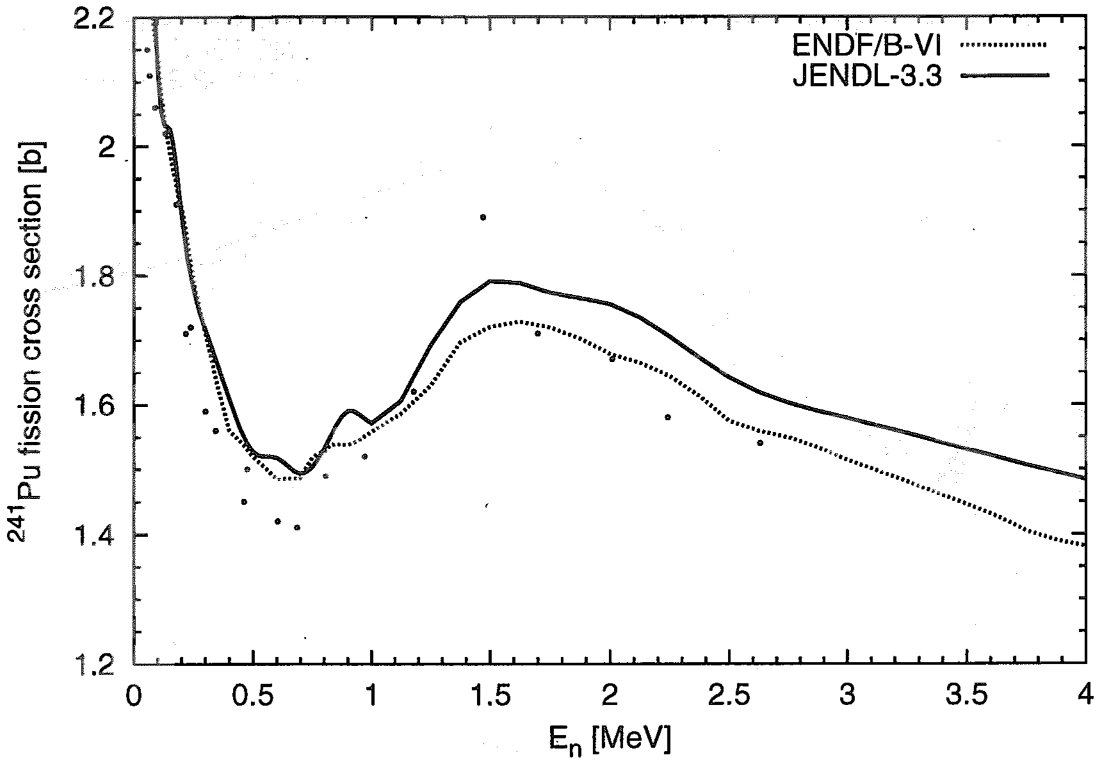

Fig. 7: ^{241}Pu fission cross sections.

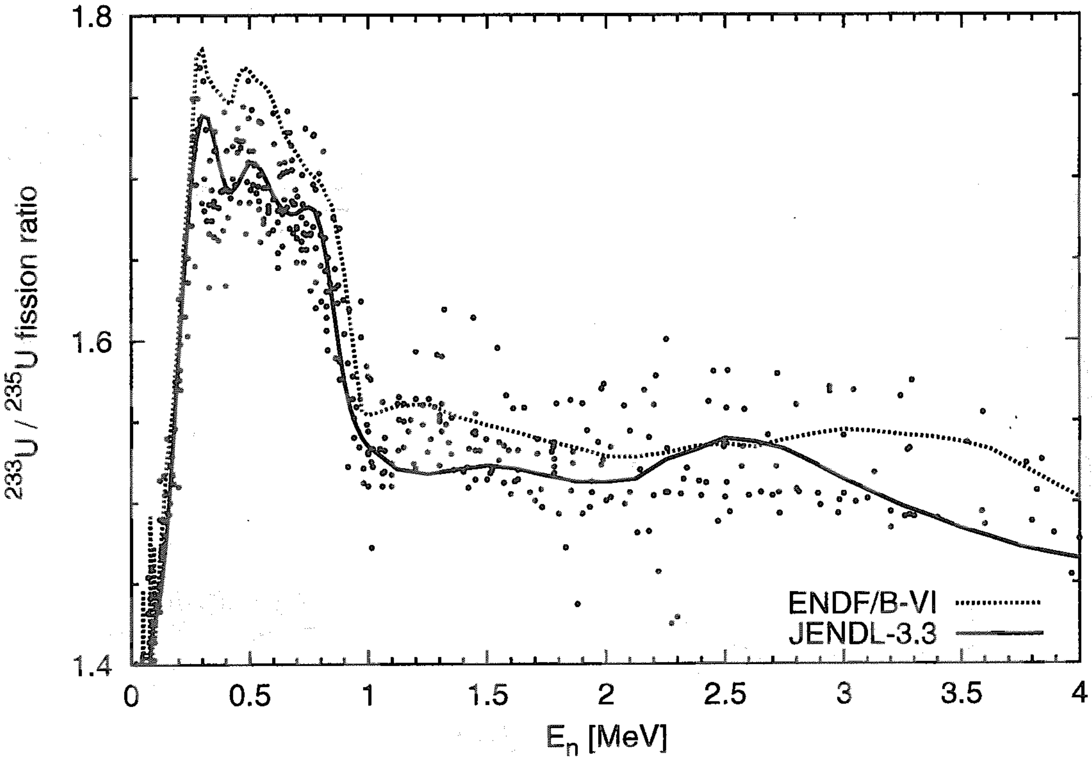

Fig. 8: Fission cross section ratio of ^{233}U to ^{235}U.

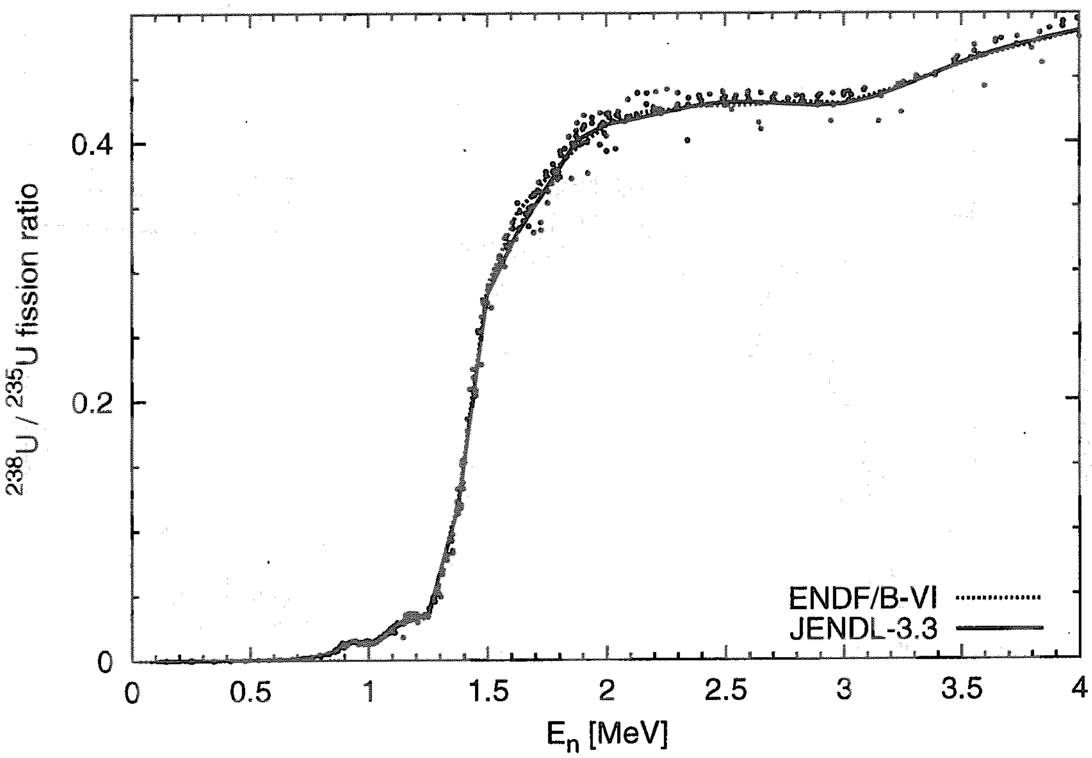

Fig. 9: Fission cross section ratio of ^{238}U to ^{235}U.

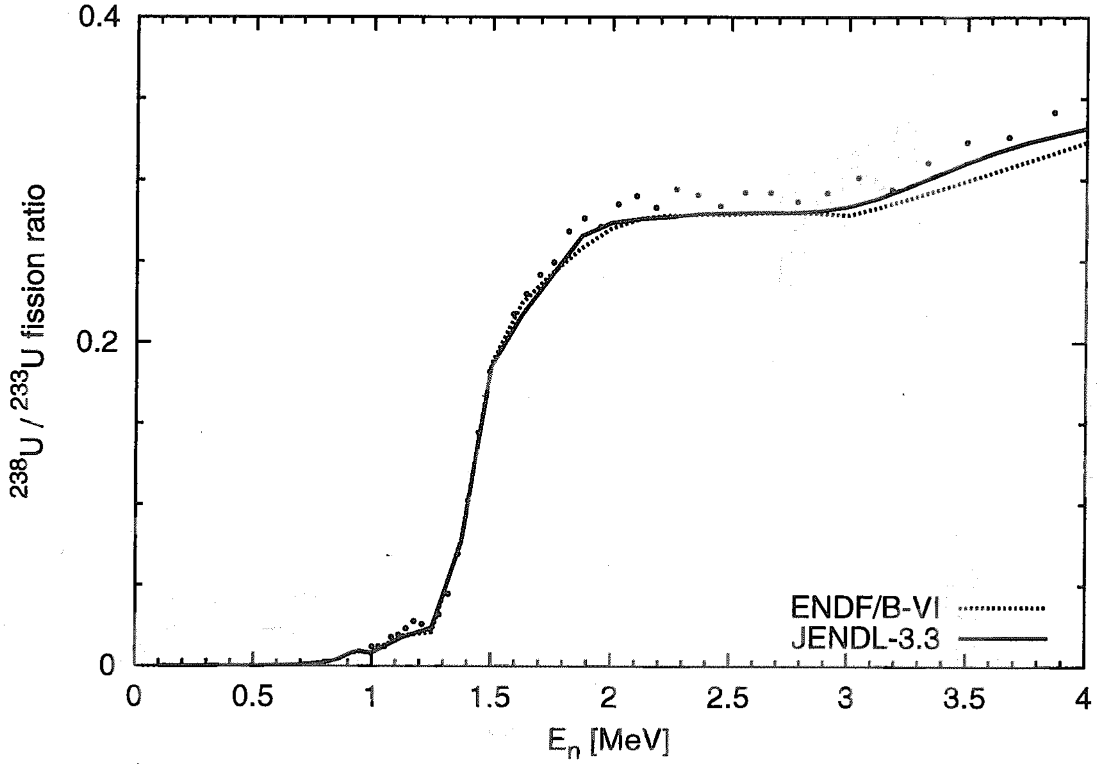

Fig. 10: Fission cross section ratio of ^{238}U to ^{233}U.

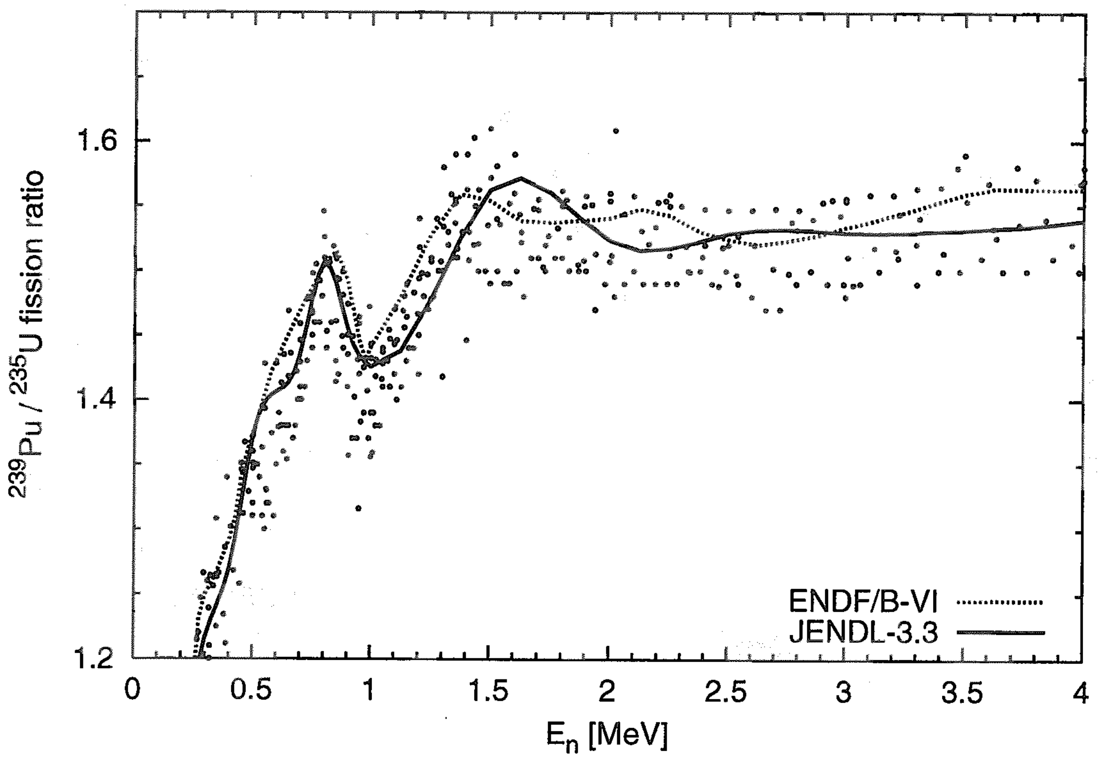

Fig. 11: Fission cross section ratio of ^{239}Pu to ^{235}U.

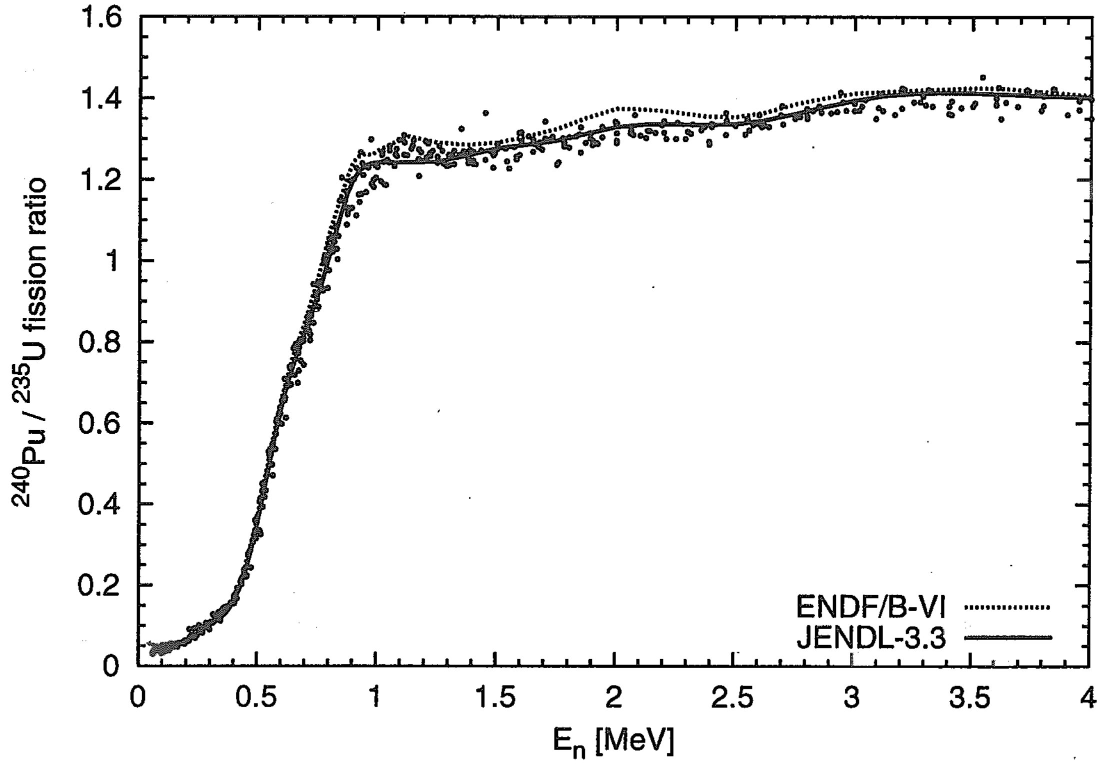

Fig. 12: Fission cross section ratio of ^{240}Pu to ^{235}U.

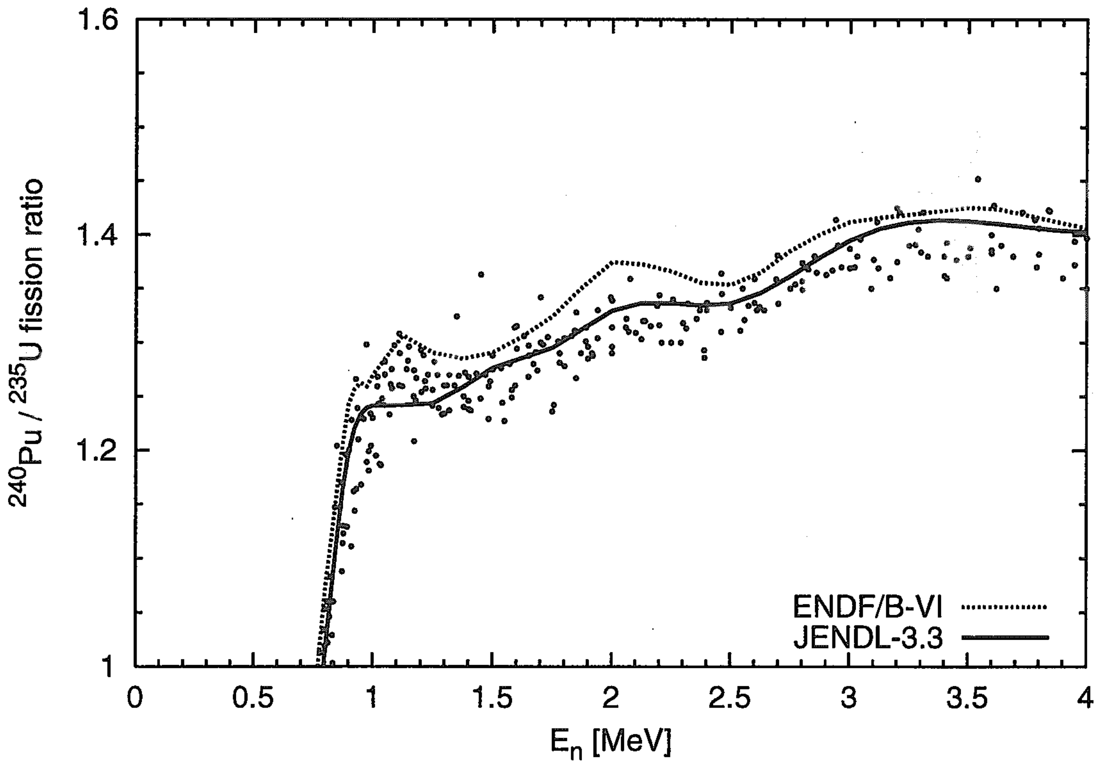

Fig. 13: Same as Fig. 12, close-up plot.

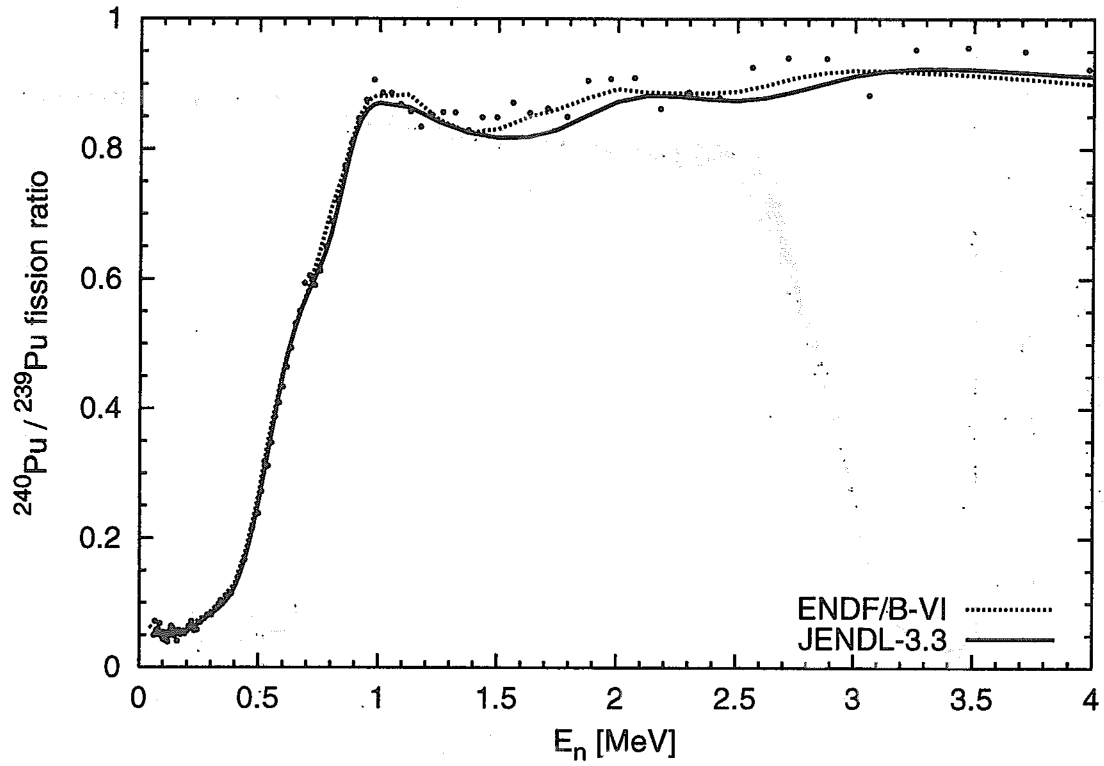

Fig. 14: Fission cross section ratio of ^{240}Pu to ^{239}Pu.

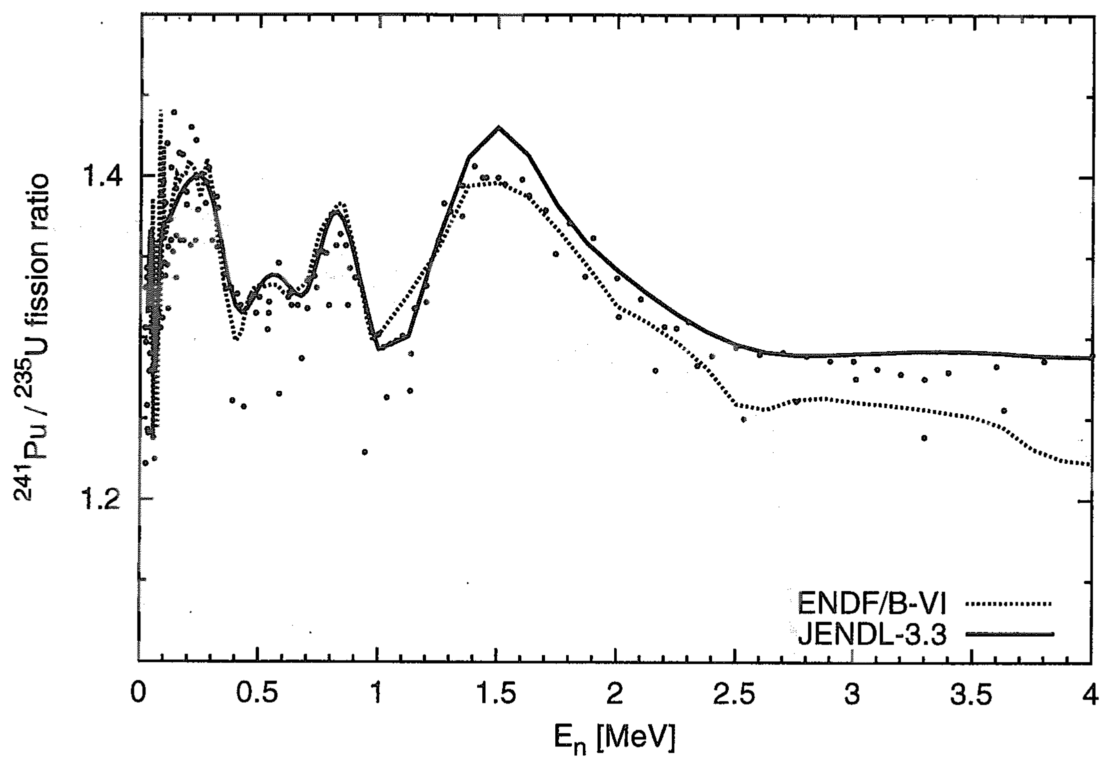

Fig. 15: Fission cross section ratio of ^{241}Pu to ^{235}U.

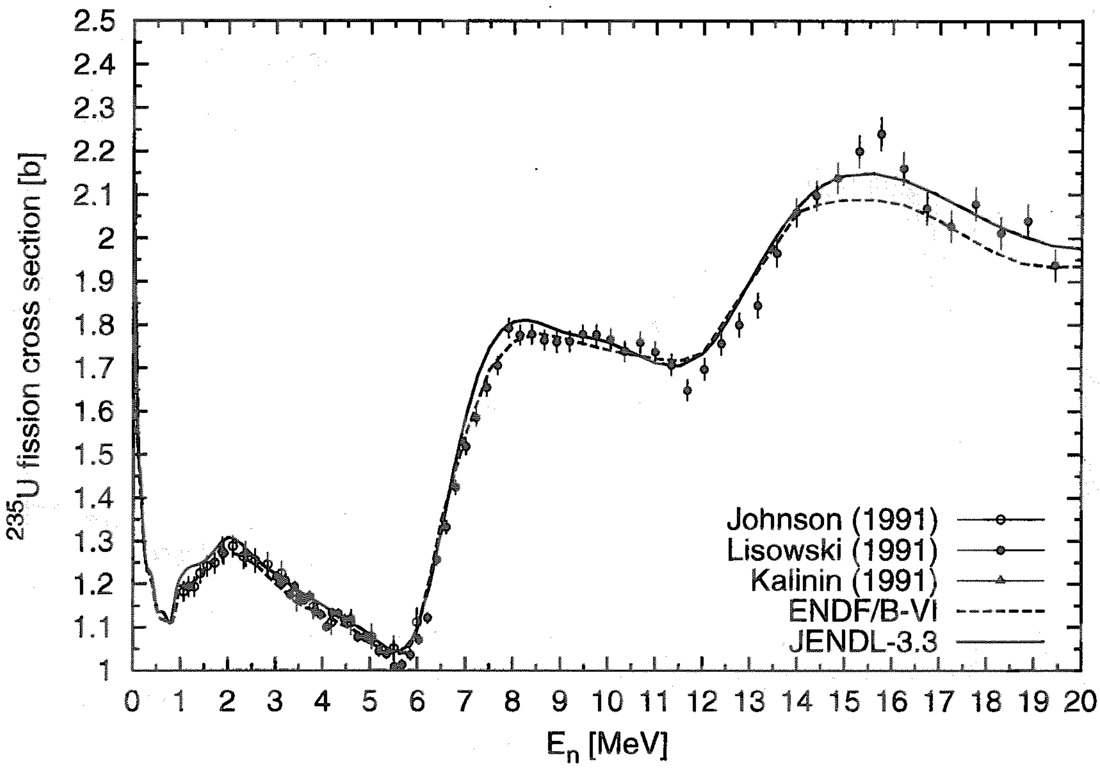

Fig. 16: Comparison of ^{235}U fission cross sections with the experimental data.

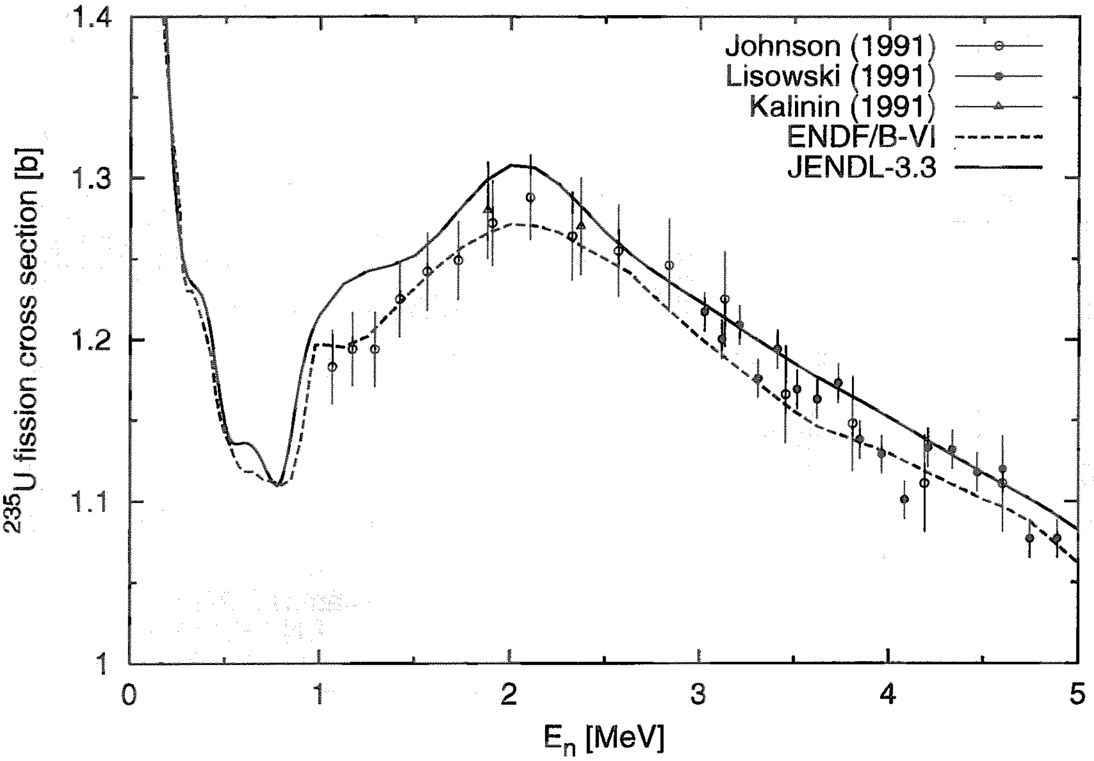

Fig. 17: Same as Fig. 16, close-up plot.

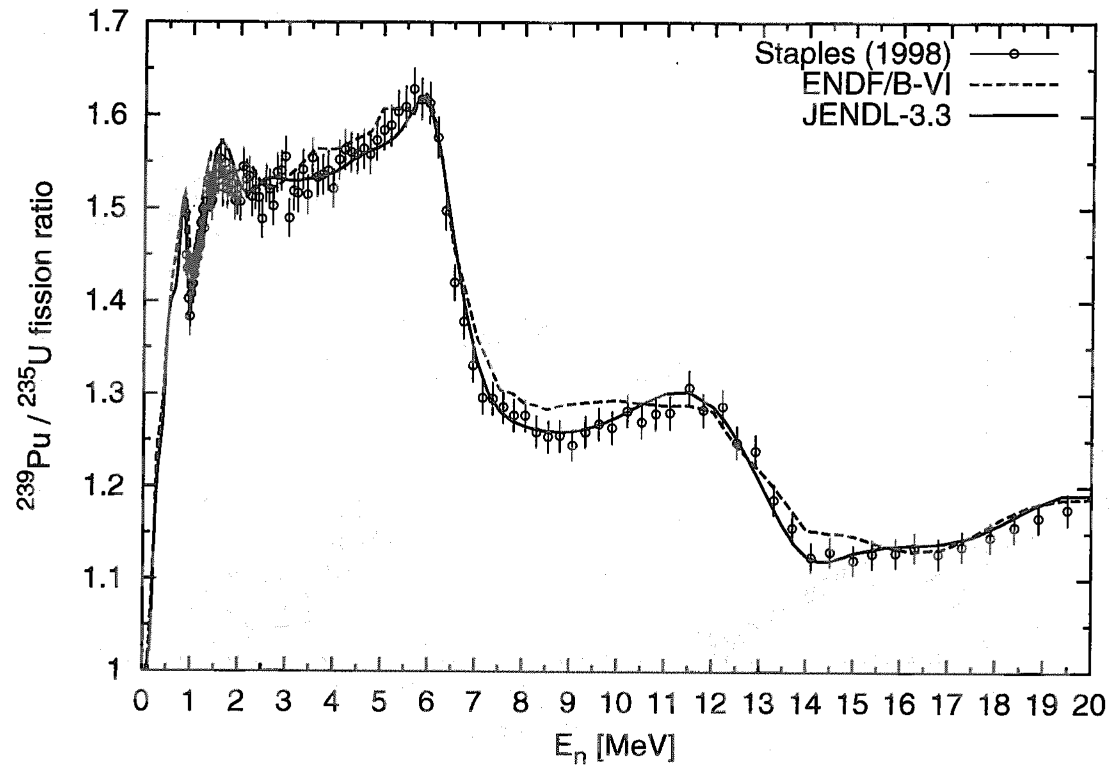

Fig. 18: Comparison of the fission cross section ratio of ^{239}Pu to ^{235}U with the experimental data of Staples.

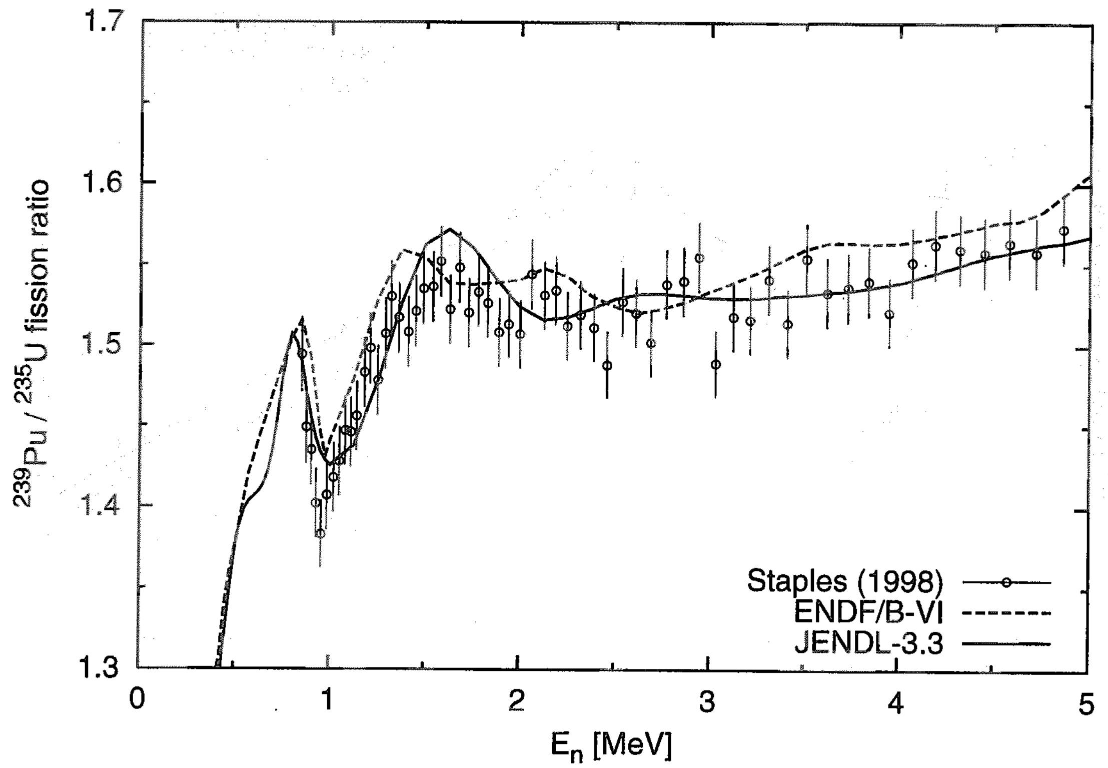

Fig. 19: Same as Fig. 18, close-up plot.

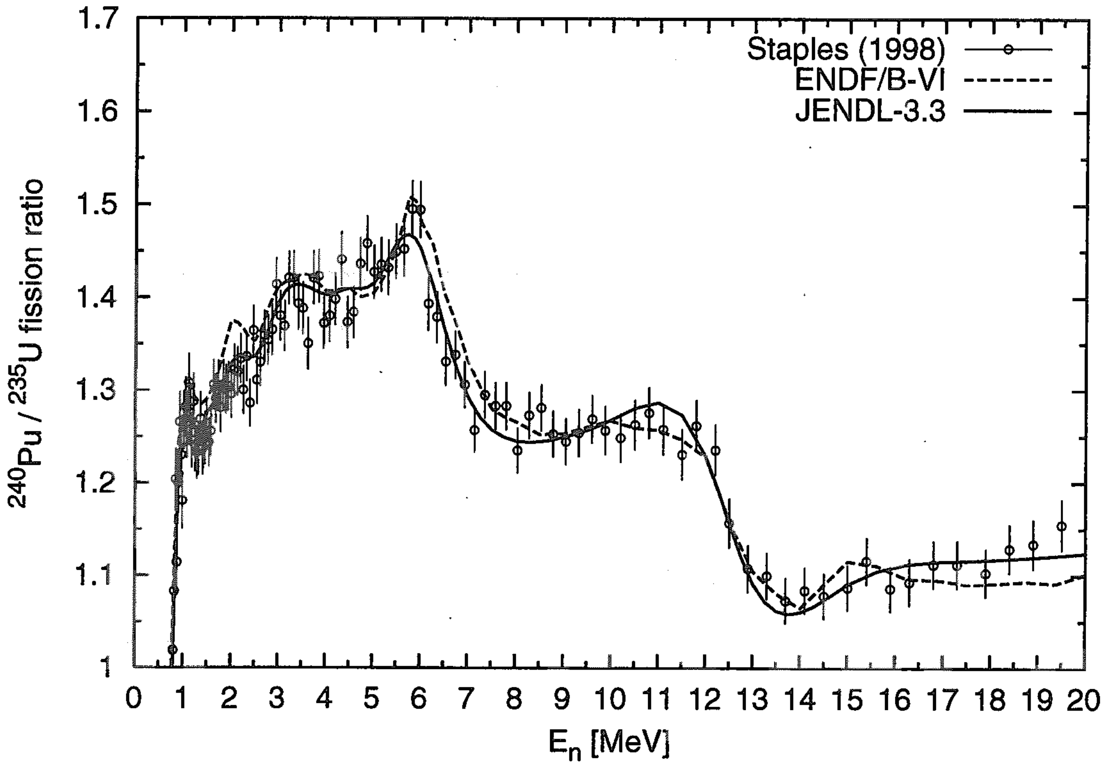

Fig. 20: Comparison of the fission cross section ratio of ^{240}Pu to ^{235}U with the experimental data of Staples.

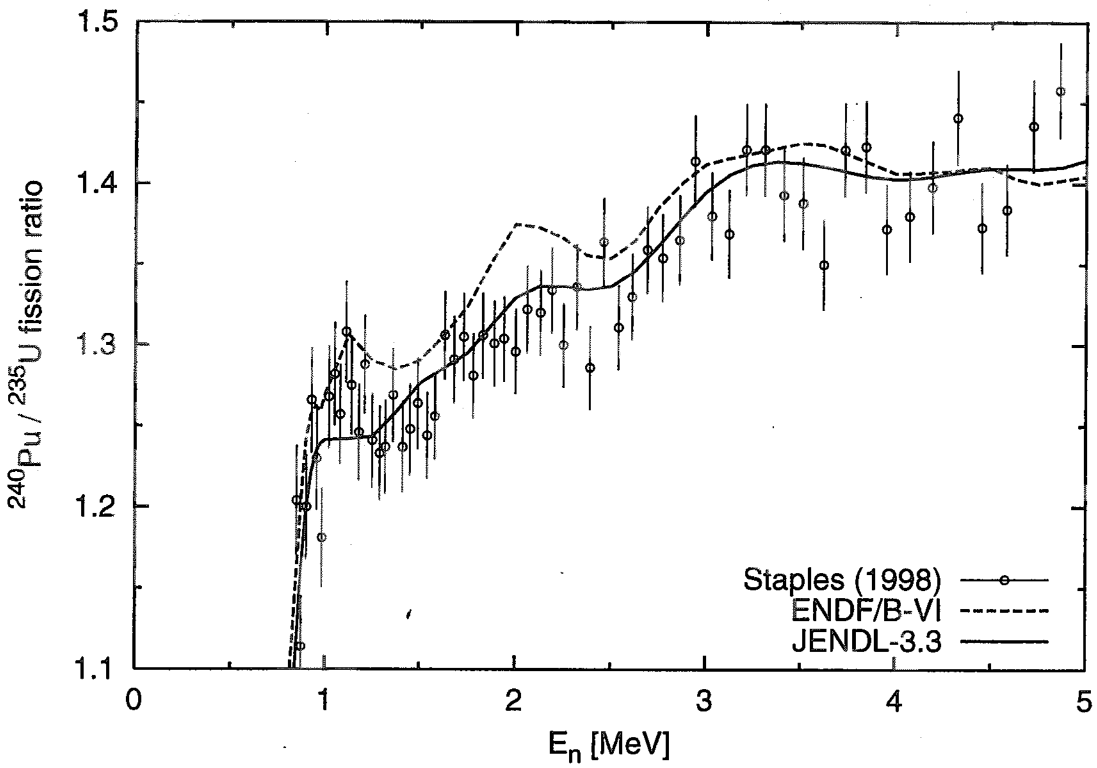

Fig. 21: Same as Fig. 18, close-up plot.

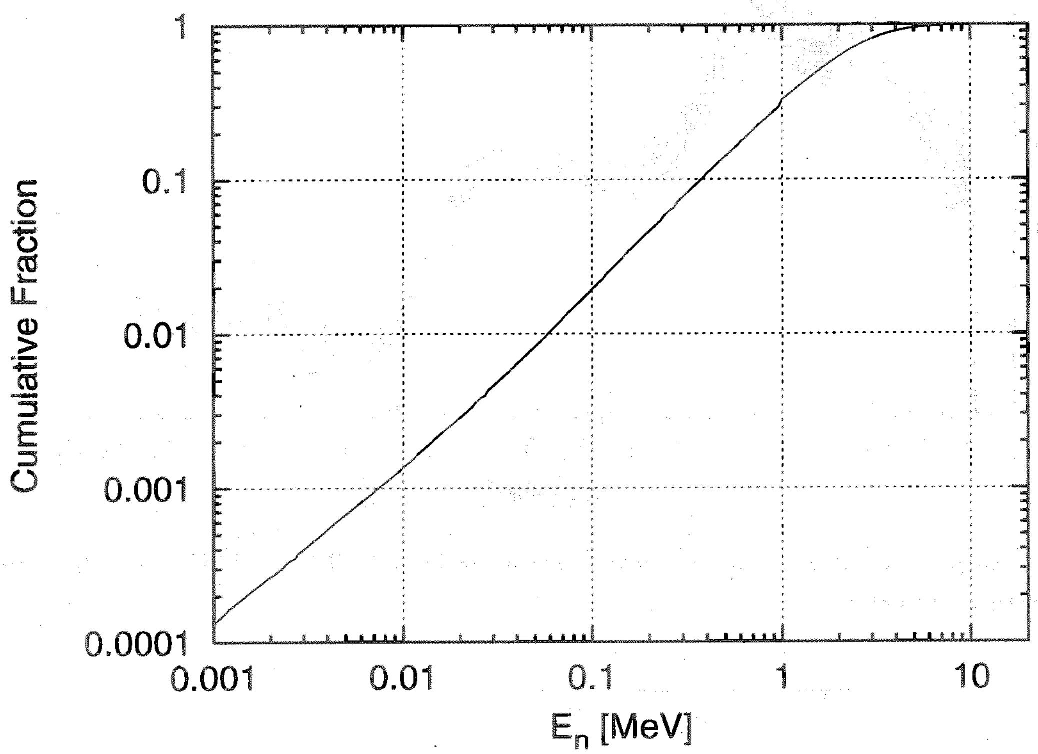

Fig. 22: Fractional contribution of the integrand $\chi(E)\sigma_f(E)$ for ^{235}U to the averaged cross section.

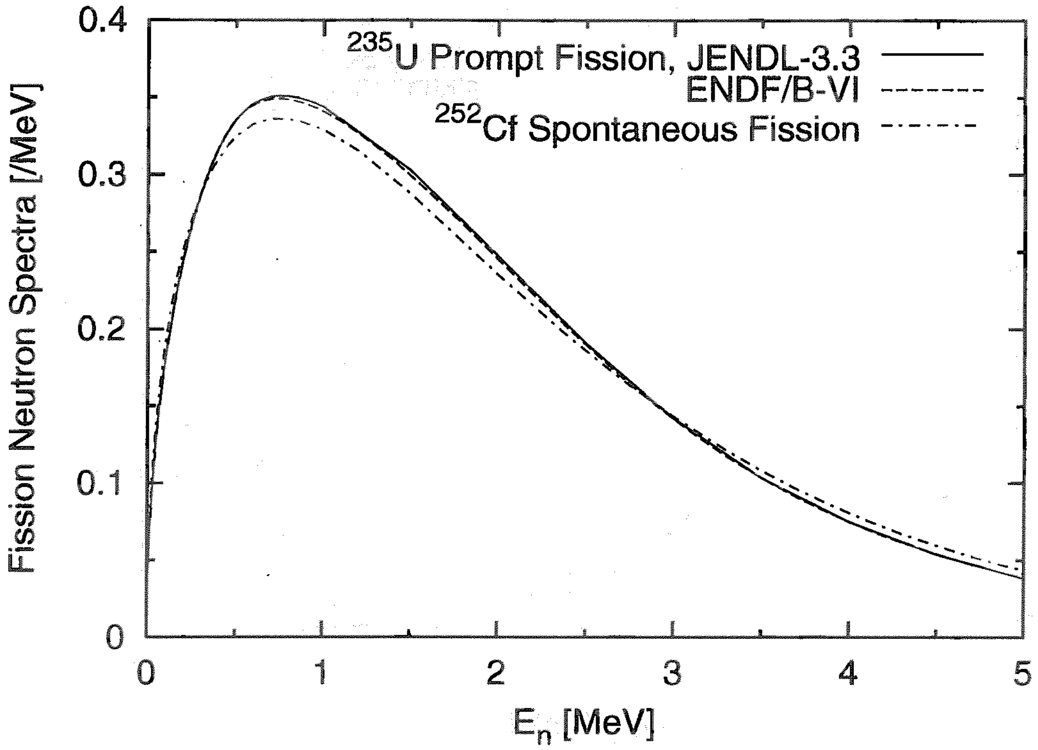

Fig. 23: Prompt fission neutron spectra $\chi(E)$ in JENDL-3.3 (solid line) and ENDF/B-VI (dot-dashed line). The dashed line represents an evaluated spectrum of ^{252}Cf spontaneous fission.

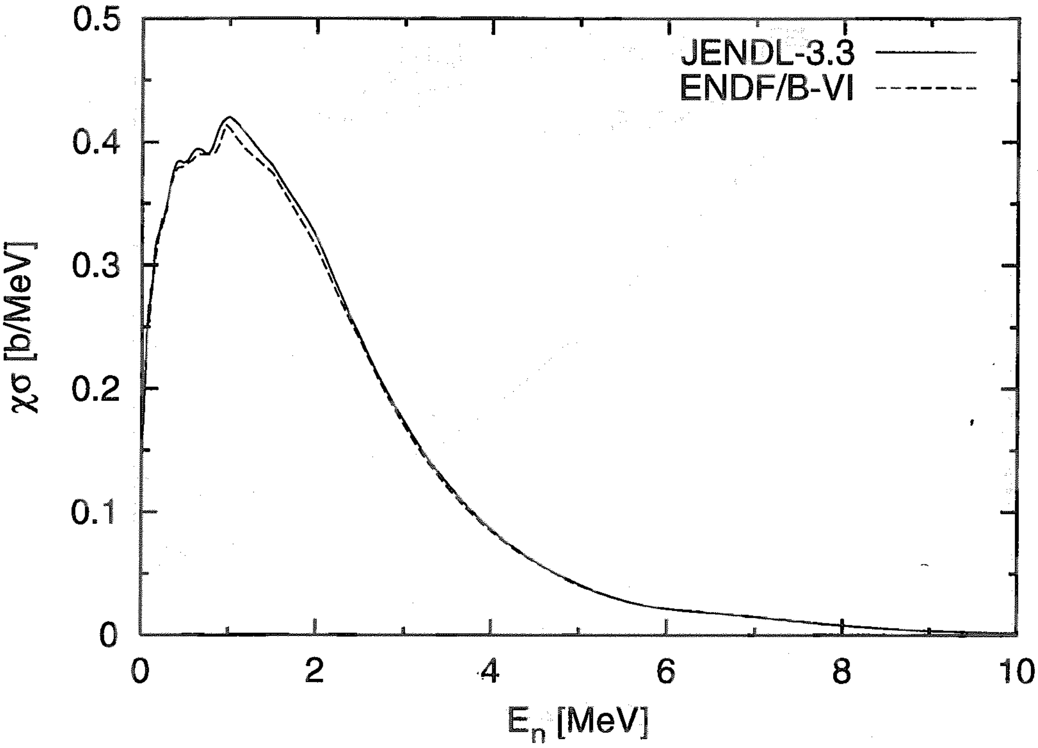

Fig. 24: Quantity $\chi(E)\sigma_f(E)$ for ^{235}U as a function of energy.

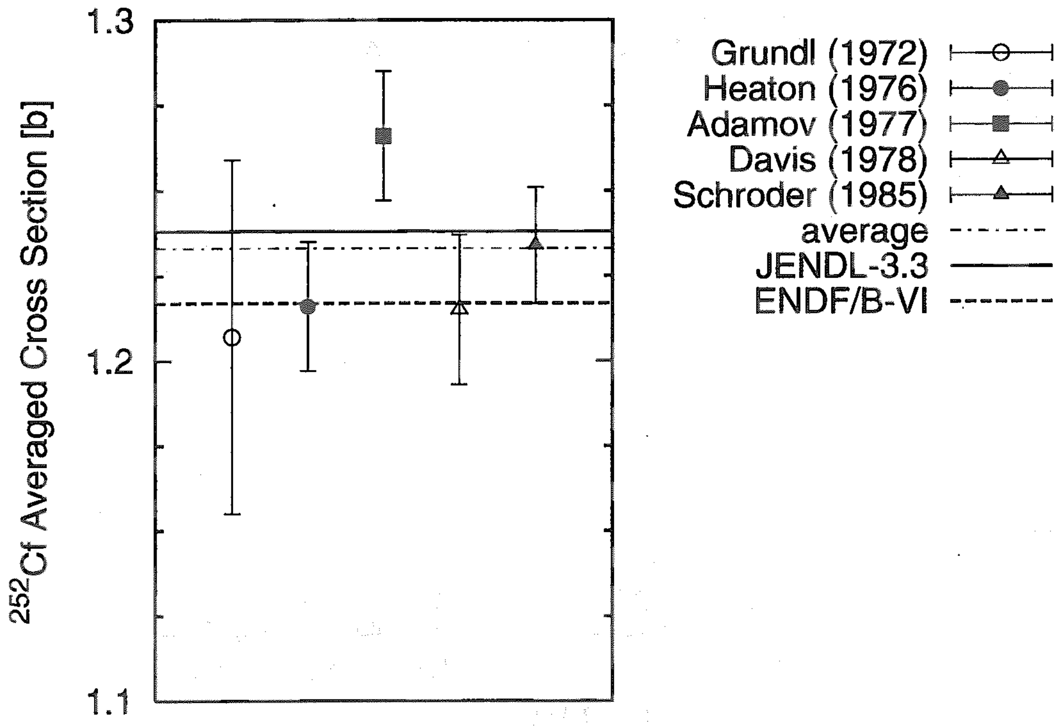

Fig. 25: Comparison of calculated ^{252}Cf spontaneous fission spectrum averaged $^{235}U(n, f)$ cross section with the experimental data. The horizontal dot-dashed line represents an averaged value of those experimental data.

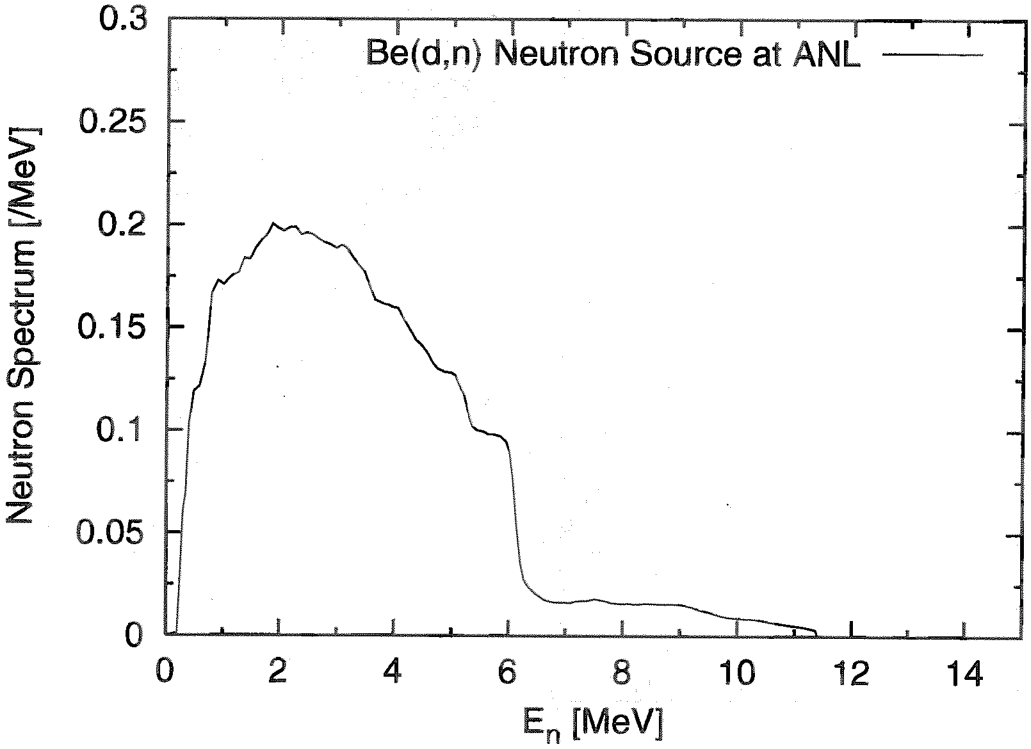

Fig. 26: Neutron spectrum produced by the ^{9}Be(d, xn) reaction.

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 eV=1.60218×10^{-19}J
1 u=1.66054×10^{-27}kg

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

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·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 Å=0.1nm=10^{-10}m
1 b=100fm^2=$10^{-28}m^2$
1 bar=0.1MPa=10^5Pa
1 Gal=1cm/s^2=10^{-2}m/s^2
1 Ci=3.7×10^{10}Bq
1 R=2.58×10^{-4}C/kg
1 rad=1cGy=10^{-2}Gy
1 rem=1cSv=10^{-2}Sv

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

粘　度　1 Pa·s(N·s/m^2)=10 P(ポアズ)(g/(cm·s))
動粘度　1m^2/s=10^4St(ストークス)(cm^2/s)

圧力	MPa(=10bar)	kgf/cm^2	atm	mmHg(Torr)	lbf/in^2(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= 4.18605J（計量法）
　= 4.184J（熱化学）
　= 4.1855J（15℃）
　= 4.1868J（国際蒸気表）

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

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）

Comparison of ^{236}U Fission Cross Sections in JENDL-3.3 and ENDF/B-VI