Annex to Memo CP-D/635

NIIAR Measurements of Fission Neutron Spectra: Summary based on Russian publications and EXFOR database

prepared by V.G. Pronyaev, edited by A. Nichols
6 November 2008

Following the development of methods to measure fission neutron spectra at the NIIAR, these measurements and publication of the results continued from the early 1970s to 1985. Many papers have been published at Kiev conferences and in Yadernye Konstanty, along with the results to be found in the EXFOR database. I am mainly considering the most recent Russian publication in Yadernye Konstanty, Vol. 3, p.16 (1985) by B.I. Starostov, V.N. Nefedov, A.A. Bojcov, as representing a full review of the measurements of this group and their final results in the form of plots. Measurement details, corrections and uncertainties, final results to be found in different EXFOR entries are given below, along with what I considered to be errors in the EXFOR compilations.

1. Measurements

The time-of-flight method with flight paths varying between 10.4 and 611 cm was used to measure fission neutron spectra over a wide energy range from 0.01 to 12 MeV. "Zero" time was the moment at which fission fragments were detected to register nuclear fission; registration of fission neutrons with time delay after the fission event gave an estimation of the energy of the neutron. ^{252}Cf and ^{235}U targets were placed in ionization chambers. A chamber with ^{235}U target was installed along the flight path of the collimated thermal neutron beam filtered from fast neutrons and gammas by 12-cm thick layer of quartz and 8-cm layer of bismuth. Different forms of detector shielding were designed for the various flight paths in order to reduce the background of scattered neutrons. Efficiency of registration of the fission fragments was above 95% for ^{252}Cf and ^{235}U over all measurement cycles. Uncertainty in "zero" time for all measurements was 0.3 nsec. Neutron detectors were installed at an angle of 45 degrees to the plane of the target in order to neutralize the spectral distortion by the non-isotropy of registration for the fission fragments.

Two measurement cycles were carried out:

Cycle No. 1 was undertaken with a miniature ionization chamber (MIC, 2.5 g) to register fission fragments, along with different flight paths and scintillation detectors. Three different series of measurements were conducted:
1.1. Flight path of 51-cm; anthracene detector; energy interval for measurements of 0.1 to 2 MeV (uncertainty in the efficiency of 2.5%).
1.2. Flight path of 231.3 cm; stilbene detector; energy interval for measurements of 1.4 to 8 MeV (uncertainty in the efficiency = uncertainty of the standard).
1.3. Flight path of 611-cm; plastic detector; interval for measurements of 3 to 12 MeV (uncertainty in the efficiency = uncertainty of the standard).

Cycle No. 2 involved gaseous scintillation detectors (GSD) for fission fragment registration, different flight paths, and two types of non-threshold neutron detector (i.e. two different sets of measurements):
2.1. Flight paths of 12.4, 21.4 and 40 cm; non-threshold ionization chambers (IC) with eight layers of ^{235}U; energy interval for measurements of 0.01 to 5 MeV (uncertainty in the efficiency less than 4%); and used only for measurements of ^{252}Cf fission neutron spectra.

2.2. Flight path of 10.4, 21.4 and 29.5 cm; gaseous scintillation detector-ionization chamber (GSDIC) with metallic ^{235}U radiator; energy interval for measurements of 0.01 to 5 MeV (uncertainty in the efficiency is determined by the uncertainty in the ^{235}U fission cross section).

Counting rates are lowest for **1.3** flight path, with 8000 counts for neutrons of energy 4 MeV, and 70 counts for neutrons with an energy of 14 MeV for 240 hours of continuous measurements.

2. Data processing

All determinations of data and their uncertainties, related to the distances, angles, numbers of counted neutrons and fission events, time channel width, and position of "zero" time, were obtained by methods described in: L.M. Green *et al*., Nucl. Sci. Eng. **50**(3) p.257 (1973); B.I. Starostov *et al*., NIIAR preprint П-12 (346) (1978)).

The effect of anisotropy on the registration of the fission fragments was studied experimentally, and was found to be negligible.

Time-of-flight spectra were transformed into energy spectra after background correction. Further corrections were introduced into these energy spectra: background neutrons scattered by the target backing, gas atoms, walls of the MIC and GSD, the air within Ω angle, the lead shielding of the detectors from the delayed-gamma and all structural parts of the neutron detector. After these corrections, the intensity ratio ^{252}Cf / ^{235}U for energy spectra in the energy interval from 0.01 to 12 MeV was obtained. This ratio is independent of the systematic uncertainties in the determination of the efficiency of the neutron detector.

Detector efficiency for **cycle No. 1** measurements was calculated for the anthracene detector by means of the Monte Carlo method, and taking into account single and double scattering of neutrons on hydrogen, nonlinearity of the photon yield for the scintillator, neutrons and gammas in the $^{12}C(n,n')$ and $^{12}C(n,n'\gamma)$ reactions, and neutrons scattered in the photo-electrical multiplier with time shift corrections. The method used to choose the threshold for neutron registration of the different detectors does not lead to discrepancies between data in the energy regions where they overlap. A calculated detector efficiency for the absolute normalization of the spectra was only used for the anthracene detectors - calculated uncertainties for the anthracene detector efficiency were below 2.5%. The efficiencies of the stilbene and plastic-scintillator detectors were determined on the supposition that the shape of the $^{252}Cf(sf)$ fission neutron spectrum is known. Such "known" spectrum was obtained by averaging (via evaluation) the data measured by many authors (see Table 2 of the paper - results are given below as taken from EXFOR in Attachment 1) - these data were used to calculate the stilbene and plastic-scintillator detector efficiencies. Efficiencies of the detectors were consistent (within 3%) with the efficiencies calculated using the Monte Carlo technique.

IC and GSDIC detector efficiencies for the **cycle No. 2** measurements of ^{252}Cf were judged to be proportional to the $^{235}U(n,f)$ cross section evaluated by V. Kon'shin and co-workers in 1978. I assume that this evaluation was inserted into the BROND-2 library for ^{235}U. Comparison of these data with ENDF/B-VII are shown at Fig. 1 of Attachment 2, and do not show any large differences in shape above a neutron energy of 15 keV. $^{235}U(n,f)$ data from BROND-2 are also given. Difficulties were experienced in the $^{235}U(n_{th},f)$ measurements with respect to background corrections (*closed geometry and large corrections? - VP*), and measurements were undertaken relative to $^{252}Cf(sf)$, with the fission neutron spectrum adopted as given in Appendix 1.

Finally, correction for the spectral resolution, suggesting that the shape of the spectra should be near-Maxwellian, was below 1.5%.

3. Uncertainties

Twenty partial components of the total uncertainty were considered. Some are uncertainties in the energy determination - those related directly to the cross-section determination are as follows:

- uncertainty in the neutron detector efficiency which is the largest partial (systematic) uncertainty
- uncertainty due to discrimination level stability
- uncertainty due to delayed gammas
- statistical uncertainty
- uncertainty due to random coincidence in the detector, which is also statistical in nature and is combined with the statistical uncertainty
- uncertainty due to "recycling" neutrons, which is also statistical in nature and is combined with statistical uncertainty
- uncertainty due to the experimental hall background, which has also statistical nature and is combined with statistical error
- uncertainty due to the flight time uncertainty
- uncertainty due to scattered neutrons.

4. Authors' conclusion

The results of the No. 1 and 2 cycles of measurements are consistent in the neutron energy range from 0.1 to 5 MeV.

Over the energy range of emitted neutrons from 0.01 to 7.5 MeV, the measured spectral shape deviates from Maxwellian by not more than 5% to 7%.

At energies above 7 MeV, there are large deviations from Maxwellian spectra.

V.G. Pronyaev: the following is my understanding (or misunderstanding) of the results from the point of view of the "primarily measured quantities" and the means of transforming to the quantities given by the authors:

1.1. series of measurements for ^{252}Cf(sf) and ^{235}U(n_{th},f) are absolute measurements. Data should be given as the number of neutrons per MeV.
1.2. series of measurements for ^{252}Cf(sf) and ^{235}U(n_{th},f) with the detector efficiency determined by means of the averaged spectrum of ^{252}Cf(sf). ^{252}Cf(sf) results for this series of measurements give only differences relative to this averaged spectrum, and ^{235}U(n_{th},f) results should exhibit the ratio to this averaged ^{252}Cf(sf) spectrum. Shape data (non-normalized) are given, and should be defined as ARB-UNITS (arbitrary). The ^{252}Cf(sf)/ ^{235}U(n_{th},f) ratio is only the directly measured quantity which does not depend from the detector efficiency determination.
1.3. series - same conclusion as for **1.2**.
2.1. series for ^{252}Cf(sf) measurements presents the results as only shape data (non-normalized) for the ratio of ^{252}Cf(sf) spectra to the ^{235}U(n,f) cross section (used as a standard for shape). Using the Kon'shin evaluation for the fission cross section, these data can be converted to shape data (non-normalized) for ^{252}Cf(sf) spectra. Data should be given as ARB-UNITS.
2.2. series - same conclusions for ^{235}U(n_{th},f) and ^{252}Cf(sf) as in the **2.1 series** for ^{252}Cf.

Table 1. Source of the latest results in EXFOR - data in EXFOR (approved by the authors in 1985-1986) compared with data at plots of final publication (YK, 3, p.16 (1985)).

Cycle Series	Reaction	EXFOR Subentry	Comment
1.1.	^{252}Cf(sf)	40874002	Data in EXFOR approved by authors, and presented as ratio to Maxwellian with kT = 1.42 MeV
	^{235}U(n_{th},f)	40871008	Data in EXFOR approved by authors in 06/1985. Since they are absolute measurements, data are given in this sub-entry as absolute spectra expressed in terms of the number of neutrons per MeV - but have been assigned incorrect ARB-UNITS in EXFOR
		40871007	Data in EXFOR approved by authors in 06/1985. Somehow normalized at a yield of 2.383 neutrons per fission, and should be given in this sub-entry as absolute spectra expressed in terms of the number of neutrons per MeV - but have been assigned incorrect ARB-UNITS in EXFOR. They are primarily obtained as shape-type data (ARB-UNITS) with efficiency of the detector evaluated using averaged ^{252}Cf(sf) fission neutron spectra
	^{252}Cf(sf)/^{235}U(n_{th},f)	40871011	Primarily measured absolute ratio, free from detector efficiency determination problems
1.2.	^{252}Cf(sf)	40871005	Data in EXFOR approved by authors in 06/1985 Somehow normalized at a yield of 3.77 neutrons per fission, and should be given in this sub-entry as absolute spectra expressed in terms of the number of neutrons per MeV - but have been assigned incorrect ARB-UNITS in EXFOR. They are primarily obtained as shape-type data (ARB-UNITS) with efficiency of the detector evaluated using averaged ^{252}Cf(sf) fission neutron spectra.
	^{235}U(n_{th},f)	40871007	Data in EXFOR approved by authors in 06/1985 Somehow normalized at a yield of 2.383 neutrons per fission, and should be given in this sub-entry as absolute spectra expressed in terms of the number of neutrons per MeV - but have been assigned incorrect ARB-UNITS in EXFOR. They are primarily obtained as shape-type data (ARB-UNITS) with efficiency of the detector evaluated using averaged ^{252}Cf(sf) fission neutron spectra
	^{252}Cf(sf)/^{235}U(n_{th},f)	40871012	Primarily measured absolute ratio, free from detector efficiency determination problems
1.3.	^{252}Cf(sf)	40872002	Data in EXFOR approved by authors in 06/1985 Somehow normalized at a yield of 3.77 neutrons per fission, and should be given in this sub-entry as absolute spectra expressed in terms of the number of neutrons per MeV - but have been assigned

			incorrect ARB-UNITS in EXFOR. They are primarily obtained as shape-type data (ARB-UNITS) with efficiency of the detector evaluated using averaged $^{252}Cf(sf)$ fission neutron spectra
	$^{235}U(n_{th},f)$	40872004	Data in EXFOR approved by authors in 06/1985 Somehow normalized at a yield of 2.383 neutrons per fission, and should be given in this sub-entry as absolute spectra expressed in terms of the number of neutrons per MeV - but have been assigned incorrect ARB-UNITS in EXFOR. They are primarily obtained as shape-type data (ARB-UNITS) with efficiency of the detector evaluated using averaged $^{252}Cf(sf)$ fission neutron spectra
	$^{252}Cf(sf)/^{235}U(n_{th},f)$	40872007	Primarily measured absolute ratio, free from detector efficiency determination problems
2.1.	$^{252}Cf(sf)$	40874003	Data in EXFOR approved by authors Data obtained with IC, and presented as ratio to Maxwellian with kT = 1.42 MeV $^{235}U(n,f)$ cross section is the standard for the shape of the $^{252}Cf(sf)$ spectra
	$^{235}U(n_{th},f)$	-	Small IC was only used for measurements of $^{252}Cf(sf)$ fission neutron spectra
2.2.	$^{252}Cf(sf)$	40874004	Data in EXFOR approved by authors Data obtained with GSDIC, and presented as ratio to Maxwellian with kT = 1.42 MeV $^{235}U(n,f)$ cross section is the standard for the shape of the $^{252}Cf(sf)$ spectra
	$^{235}U(n_{th},f)$	40873004	Data in EXFOR approved by authors Data are presented as ratio to Maxwellian with kT = 1.313 MeV Shape-type data, with the efficiency of the detector evaluated using averaged $^{252}Cf(sf)$ fission neutron spectra

Finally, the following data sets can be included in the least-squares fit and evaluation:

40874002 are the result of absolute measurements (but could be used as shape-type data) - neutron detector correlation with 40871008.

40871008 are the result of absolute measurements (but could be used as shape-type data) - neutron detector correlation with 40871002.

40871011 are the result of direct measurement of the normalized ratio (but could be used as shape-type data for ratio). If 40871011 is used, 40874002 and 40871008 should not be used (to avoid duplication and redundancy).

40871012 (with two data sets) are the result of direct measurement of normalized ratio (but could be used as shape-type data for ratio).

40872007 (with two datasets) are the result of direct measurement of normalized ratio (but could be used as shape-type data for ratio).

40874003 is measurement relative to the ^{235}U(n,f) cross section. Can be renormalized to the new ^{235}U(n,f) cross-section standard. Possess a common standard, and therefore have correlations with 40874004 and 40873004.

40874004 is measurement relative to the ^{235}U(n,f) cross section. Can be renormalized to the new ^{235}U(n,f) cross-section standard. Possess a common standard, and therefore have correlations with 40874003 and 40873004.

If more stringent conditions are adopted for selection, as proposed by Nikolai Kornilov (direct measurements and more accurate ratios for ^{235}U to ^{252}Cf spectra), only EXFOR sub-entries 40871011, 40871012 and 40872007 are appropriate for consideration as data for standard and reference spectra fit.

Table 2. Other data in the EXFOR database (data presented – most of my guesses are based on information given by the compiler).

EXFOR Sub-entry	**Data presented**
40871007	Data derived by combining the results of **1.1**. and **1.2. series** of measurements (given as a single data set, without the inclusion of overlapping data), and presented as absolute data (number of neutrons/MeV) There is no description of the procedure in REFERENCE – they were probably sent by authors to the compiling centre as additional data for compilation
40644002	Data derived by combining the results of all series of measurements (as a single dataset, without the inclusion of overlapping data), and presented as absolute data in the energy range from 14.3 keV to 10.14 MeV (number of neutrons per MeV) There is no explanation of how the data were combined

Table 3. Errors and misprints found in the current EXFOR database.

EXFOR Sub-entry	**Important problem**
40873001	MONITOR is the spectrum given in Table 4 of YK, vol. 3, p.20 (1985) (see also Attachment 1 or X4 = 40930002), which is presented as the ratio to the Maxwellian spectra with kT = 1.42 MeV, and not Maxwellian spectra with kT = 1.418
40871008	REACTION: not relative to the Maxwellian
40871005	REACTION: should be (sf) and not (n,f) Data units are probably the number of neutrons per MeV
40871007	Data units should be the number of neutrons per MeV
40871008	Data units should be the number of neutrons per MeV
40872001	DETECTOR (SCIN) free text should be "plastic scintillator for neutron spectrum measurements"
40872002	Data are given as number of neutrons per MeV, although they are shape-type data
40872004	Data are given as number of neutrons per MeV, although they are shape-type data
40873001	MONITOR is the spectrum given in Table 4 of YK, vol. 3, p.20 (1985) (see also Attachment 1 or X4 = 40930002), which is presented as the ratio to the Maxwellian spectra with kT = 1.42 MeV, and not Maxwellian spectra with kT = 1.418
40873004	REACTION: should be relative Maxwellian with kT = 1.313 MeV

40874002	Data units should be the number of neutrons per MeV – absolute measurements
40874001	DETECTOR: (SCIN) anthracene detector for neutron registration should be moved to 40874002
40874003	DETECTOR should be "(IOCH) thin-wall ionization chamber with eight ^{235}U fission layer for neutron registration"
40874003	DETECTOR should be "(IOCH) gas scintillating detector - ionization chamber with ^{235}U metallic radiator"
40930002	Data are the result of much experimental data averaging used to determine the efficiency of the detectors - they should be inserted (as evaluated data - monitor) in all subentries of **cycles 1.2. and 1.3**

Attachment 1. ^{252}Cf(sf) fission neutron spectra used to determine the absolute efficiency of the stilbene (case 1.2. for cycle 1) and plastic detectors (case 1.3. for cycle 1) given as ratio to the Maxwellian spectrum with kT = 1.42 MeV.

```
SUBENT        40930002   19990311   19990705   20050926       000040930002    1
BIB                  4         12                                 40930002    2
REACTION   ((98-CF-252(0,F),PR,DE,N,,EXP)//                       40930002    3
           (98-CF-252(0,F),PR,DE,N,,CALC))  RELATIVE TO THE       40930002    4
              MAXWELL SPECTRUM WITH TEMPERATURE 1.42 MEV          40930002    5
STATUS     (TABLE)  DATA ARE TAKEN FROM TABLE 4 OF MAIN REFERENCE 40930002    6
           (COREL,40874003)  DATA IN THIS SUBENT COVER THE WHOLE  40930002    7
           (COREL,40874004)  ENERGY RANGE IN DIFFERENCE TO SUBENT 40930002    8
                             40874002 AND 40874004                40930002    9
FLAG       (1.)  INDEPENDENT VARIABLE CHANGED BY COMPILER         40930002   10
HISTORY    (19990311A)  REACTION VALUE IS GIVEN EXPLICITELY AS    40930002   11
                        RATIO OF TWO VALUES                       40930002   12
           (19990405A)  NEW REACTION-QUANTITY RATIOS CODING GIVEN 40930002   13
                        E-NM AND E-DN INTRODUCED                  40930002   14
ENDBIB              12                                            40930002   15
COMMON               1          3                                 40930002   16
E-DN                                                              40930002   17
MEV                                                               40930002   18
  1.42                                                            40930002   19
ENDCOMMON            3                                            40930002   20
DATA                 5        110                                 40930002   21
E-NM       E-RSL      DATA       ERR-T      FLAG                  40930002   22
MEV        MEV        NO-DIM     PER-CENT   NO-DIM                40930002   23
  3.000E-04  1.0  E-04  1.084E-00  40.                            40930002   24
  7.000E-04  2.0  E-04  1.084E-00  30.                            40930002   25
  1.500E-03  4.0  E-04  9.58 E-01  20.                            40930002   26
  2.500E-03  3.0  E-04  9.58 E-01  15.                            40930002   27
  3.400E-03  3.0  E-04  9.72 E-01  10.                            40930002   28
  4.4  E-03  3.0  E-04  9.72 E-01  10.                            40930002   29
  5.5  E-03  3.0  E-04  9.57 E-01   9.                            40930002   30
  6.5  E-03  3.0  E-04  9.55 E-01   8.                            40930002   31
  7.5  E-03  3.0  E-04  9.83 E-01   8.                            40930002   32
  8.4  E-03  4.0  E-04  9.83 E-01   8.                            40930002   33
  9.5  E-03  6.0  E-04  9.75 E-01   7.                            40930002   34
  1.540E-02  2.6  E-03  9.73 E-01   7.                            40930002   35
  3.440E-02  2.1  E-03  9.65 E-01   7.                            40930002   36
  6.000E-02  2.00 E-03  9.71 E-01   6.                            40930002   37
  8.100E-02  3.0  E-03  9.88 E-01   5.                            40930002   38
  9.100E-02  3.0  E-03  1.000E-00   4.                            40930002   39
  1.040E-01  3.0  E-03  1.028E-00   3.                            40930002   40
  1.150E-01  3.0  E-03  1.026E-00   3.                            40930002   41
  1.340E-01  3.0  E-03  1.010E-00   2.5                           40930002   42
  1.440E-01  3.0  E-03  1.018E-00   2.5                           40930002   43
  1.550E-01  3.0  E-03  9.99 E-01   2.5                           40930002   44
  1.660E-01  3.0  E-03  9.83 E-00   2.5                           40930002   45
  1.780E-01  3.0  E-03  9.70 E-01   2.5                           40930002   46
  1.940E-01  4.0  E-03  9.62 E-01   2.5                           40930002   47
  2.070E-01  4.0  E-03  9.64 E-01   2.5                           40930002   48
  2.210E-01  4.0  E-03  9.68 E-01   2.5                           40930002   49
  2.390E-01  5.0  E-03  9.75 E-01   3.                            40930002   50
  2.630E-01  6.0  E-03  9.74 E-01   3.                            40930002   51
  2.87 E-01  6.0  E-03  9.89 E-01   3.                            40930002   52
  3.20 E-01  6.0  E-03  9.82 E-01   3.                            40930002   53
  3.30 E-01  7.0  E-03  9.89 E-01   3.        1.                  40930002   54
  3.57 E-01  8.0  E-03  9.88 E-01   2.                            40930002   55
  3.82 E-01  8.0  E-03  9.82 E-01   2.                            40930002   56
  4.27 E-01  8.0  E-03  9.89 E-01   2.                            40930002   57
  4.56 E-01  8.0  E-03  9.73 E-01   2.                            40930002   58
  4.99 E-01  8.0  E-03  9.76 E-01   2.                            40930002   59
  5.49 E-01  8.0  E-03  9.79 E-01   2.                            40930002   60
  6.31 E-01  6.0  E-03  9.62 E-01   2.                            40930002   61
  6.92 E-01  6.0  E-03  9.76 E-01   2.                            40930002   62
  7.37 E-01  6.0  E-03  9.77 E-01   2.                            40930002   63
  8.03 E-01  7.0  E-03  9.73 E-01   2.                            40930002   64
  8.68 E-01  9.0  E-03  9.78 E-01   2.                            40930002   65
  9.06 E-01  1.1  E-02  1.000E-00   2.                            40930002   66
  1.002E-00  2.8  E-02  9.96 E-01   2.                            40930002   67
  1.050E-00  2.8  E-02  1.002E-01   2.                            40930002   68
  1.170E-00  2.8  E-02  1.011E-01   2.                            40930002   69
  1.260E-00  2.9  E-02  1.023E-01   3.                            40930002   70
  1.353E-00  3.5  E-02  1.028E-01   2.5                           40930002   71
```

1.480E-00	2.9	E-02	1.044E-01	2.5	40930002	72
1.640E-00	2.6	E-02	1.026E-01	2.5	40930002	73
1.760E-00	2.5	E-02	1.020E-01	2.5	40930002	74
1.836E-00	2.6	E-02	1.027E-01	2.	40930002	75
1.990E-00	3.0	E-02	1.024E-01	2.	40930002	76
2.123E-00	3.1	E-02	1.023E-01	2.	40930002	77
2.216E-00	3.1	E-02	1.027E-01	1.5	40930002	78
2.314E-00	3.1	E-02	1.024E-01	1.5	40930002	79
2.400E-00	3.1	E-02	1.030E-01	1.5	40930002	80
2.537E-00	3.8	E-02	1.034E-01	2.	40930002	81
2.662E-00	3.3	E-02	1.030E-01	2.2	40930002	82
2.772E-00	3.0	E-02	1.034E-01	2.5	40930002	83
2.875E-00	4.1	E-02	1.026E-01	2.6	40930002	84
2.964E-00	3.1	E-02	1.034E-01	2.2	40930002	85
3.151E-00	3.4	E-02	1.020E-01	2.	40930002	86
3.305E-00	3.3	E-02	1.024E-01	2.5	40930002	87
3.408E-00	6.2	E-02	1.015E-01	2.3	40930002	88
3.537E-00	6.2	E-02	1.015E-01	2.3	40930002	89
3.629E-00	6.8	E-02	1.014E-01	2.3	40930002	90
3.748E-00	6.8	E-02	1.014E-01	2.3	40930002	91
3.938E-00	6.2	E-02	1.017E-01	2.3	40930002	92
4.155E-00	6.2	E-02	1.015E-01	2.5	40930002	93
4.268E-00	6.2	E-02	1.015E-01	2.5	40930002	94
4.398E-00	6.5	E-02	1.017E-01	2.3	40930002	95
4.582E-00	6.5	E-02	1.015E-01	2.5	40930002	96
4.777E-00	6.5	E-02	1.015E-01	2.	40930002	97
4.986E-00	6.5	E-02	1.013E-01	2.	40930002	98
5.208E-00	6.5	E-02	1.018E-01	2.	40930002	99
5.446E-00	6.5	E-02	1.011E-01	3.	40930002	100
5.700E-00	6.5	E-02	9.99 E-01	4.	40930002	101
5.973E-00	6.5	E-02	9.93 E-01	4.	40930002	102
6.170E-00	1.5	E-01	9.89 E-01	4.	40930002	103
6.270E-00	1.5	E-01	9.88 E-01	4.	40930002	104
6.370E-00	1.6	E-01	9.88 E-01	4.	40930002	105
6.470E-00	1.6	E-01	9.89 E-01	4.	40930002	106
6.580E-00	1.6	E-01	9.87 E-01	4.	40930002	107
6.69	1.6	E-01	9.85 E-01	4.	40930002	108
6.81	1.6	E-01	9.89 E-01	5.	40930002	109
6.92	1.6	E-01	9.86 E-01	5.	40930002	110
7.04	1.6	E-01	9.86 E-01	5.	40930002	111
7.16	1.6	E-01	9.84 E-01	5.	40930002	112
7.29	1.6	E-01	9.80 E-01	5.	40930002	113
7.42	2.0	E-01	9.74 E-01	5.	40930002	114
7.55	2.0	E-01	8.69 E-01	5.	40930002	115
7.68	2.0	E-01	9.59 E-01	5.	40930002	116
7.82	2.0	E-01	9.52 E-01	5.	40930002	117
7.97	2.0	E-01	9.50 E-01	5.5	40930002	118
8.11	2.0	E-01	9.49 E-01	5.5	40930002	119
8.27	2.0	E-01	9.46 E-01	6.	40930002	120
8.42	2.0	E-01	9.43 E-01	6.	40930002	121
8.58	2.0	E-01	9.41 E-01	6.	40930002	122
8.75	2.0	E-01	9.32 E-01	6.	40930002	123
9.09	2.0	E-01	9.23 E-01	6.	40930002	124
9.46	2.0	E-01	9.15 E-01	6.5	40930002	125
9.92	2.0	E-01	9.27 E-01	6.	40930002	126
10.0	2.0	E-01	9.06 E-01	6.5	40930002	127
10.7	2.0	E-01	8.73 E-01	7.	40930002	128
11.3	2.0	E-01	8.50 E-01	7.	40930002	129
11.9	2.0	E-01	8.38 E-01	7.	40930002	130
12.5	2.0	E-01	8.31 E-01	8.	40930002	131
13.6	2.	E-01	7.89 E-01	9.	40930002	132
15.4	3.	E-01	7.72 E-01	10.	40930002	133

Attachment 2. ^{235}U fission cross sections used to calculate the energy dependence of the detector efficiency in cycle No. 2 measurements.

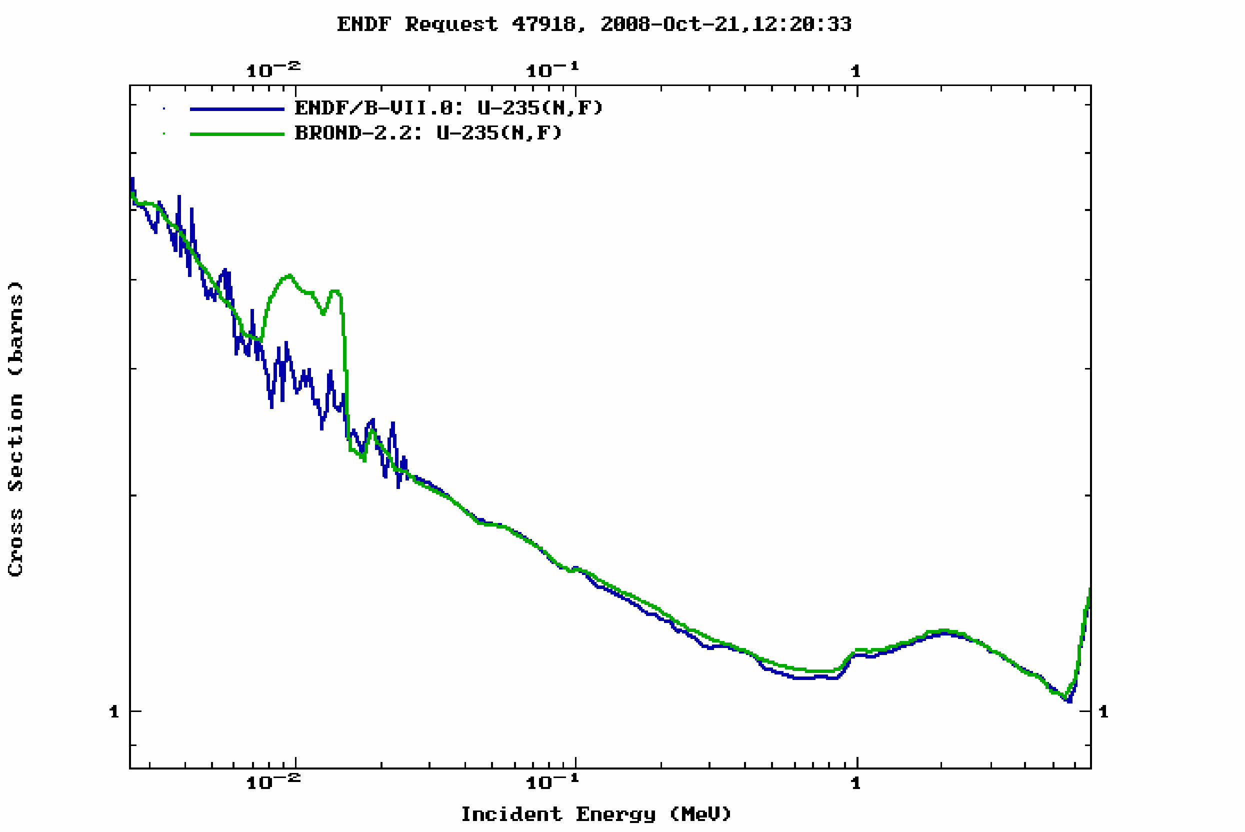

Fig. 1. Comparison of Kon'shin evaluation (1978) with ENDF/B-VII.0.

Table 1. Cross sections of $^{235}U(n,f)$ used to evaluate the energy dependence of the efficiency of the detectors in cycle 2 measurements.

#	Energy	Cross section
#	MeV	barns
	0.0100437	3.94808
	0.0103271	3.88585
	0.0105	3.84787
	0.0107963	3.84031
	0.0111009	3.83253
	0.0114141	3.82454
	0.0115	3.82235
	0.0118245	3.74199
	0.012	3.69853
	0.0123386	3.62146
	0.0125	3.58472
	0.0128527	3.68319
	0.013	3.72432
	0.0133668	3.81562
	0.0135	3.84878
	0.013625	3.85819
	0.0136875	3.86029
	0.01375	3.86085
	0.013875	3.85771
	0.014	3.84941
	0.014125	3.83633
	0.01425	3.81864

0.0143125	3.80806
0.0145	3.76912
0.01475	3.43044
0.014875	3.25625
0.015	3.07821
0.015125	2.89572
0.01525	2.70809
0.015375	2.51446
0.0155	2.31375
0.0159374	2.31238
0.016	2.31219
0.0164515	2.29604
0.0165	2.2943
0.0169656	2.27187
0.017	2.27021
0.0174797	2.24546
0.0175	2.24441
0.017625	2.28273
0.01775	2.31678
0.017875	2.34742
0.018	2.37527
0.01825	2.42437
0.0185	2.46671
0.01875	2.45287
0.019	2.43264
0.019125	2.42042
0.01925	2.40687
0.0195	2.37595
0.0200502	2.35068
0.02025	2.3415
0.020625	2.32138
0.021	2.29889
0.021375	2.27358
0.02175	2.24483
0.022125	2.21181
0.0225	2.17334
0.023125	2.1719
0.02375	2.16595
0.024375	2.15661
0.025	2.14467
0.0257054	2.12799
0.02625	2.11511
0.0269907	2.09447
0.0275	2.08028
0.028276	2.06855
0.0290738	2.0565
0.029375	2.05195
0.0302039	2.0412
0.0310561	2.03014
0.03125	2.02762
0.0321318	2.01843
0.0330384	2.00897
0.0339706	1.99925
0.0349292	1.98925
0.035	1.98851
0.0359876	1.97198
0.037003	1.95497
0.0380471	1.93749
0.0391207	1.91952
0.04	1.90479
0.0411287	1.88854

0.0422892	1.87184
0.0434824	1.85466
0.0447093	1.837
0.045	1.83281
0.0462697	1.8283
0.0475	1.82394
0.0488403	1.82071
0.05	1.81792
0.0514108	1.81638
0.0528615	1.8148
0.054353	1.81317
0.055	1.81246
0.0565519	1.80014
0.0581476	1.78747
0.0597883	1.77444
0.06	1.77276
0.061693	1.76098
0.0634338	1.74888
0.065	1.73798
0.0668341	1.72844
0.0687199	1.71862
0.07	1.71195
0.0719752	1.70289
0.074006	1.69356
0.075	1.689
0.0771162	1.67077
0.0792922	1.65202
0.08	1.64592
0.0822573	1.62774
0.0845783	1.60905
0.085	1.60566
0.0873984	1.59751
0.0898645	1.58913
0.0924001	1.58051
0.095	1.57168
0.1	1.57775
0.1	1.58098
0.126764	1.50533
0.152116	1.45355
0.223104	1.33876
0.25	1.302
0.29	1.267
0.443157	1.1877
0.47298	1.17578
0.514108	1.1639
0.608465	1.14432
0.709876	1.137
0.74503	1.137
0.8	1.139
0.85	1.147
0.9	1.168
0.95	1.202
1	1.22
1.1	1.215
1.23386	1.22322
1.45981	1.24633
1.8	1.288
2	1.298
2.23104	1.28983
2.46772	1.27157
2.98012	1.22109

3.90722	1.13942
4	1.132
4.5	1.111
5	1.064
5.5	1.047
6	1.11201
6.5	1.364
7	1.553
7.5	1.719
8	1.782
8.47892	1.782
10.0437	1.74895
10.6481	1.73622
11.1552	1.732
11.4052	1.732
11.5	1.732
12	1.748
12.5	1.826
13.1834	1.94545
13.5	1.998
14	2.068
14.5	2.099
15	2.103
16	2.068
17	1.986
18	1.939
19	1.966
20	2.045