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NUCLEAR DATA STANDARDS

FOR NUCLEAR MEASUREMENTS

1991 NEANDC/INDC
NUCLEAR STANDARDS FILE

Editor

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NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

1992

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

Pursuant to Article 1 of the Convention signed in Paris on 14th December 1960, and which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development (OECD) shall promote policies designed:

— to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and thus to contribute to the development of the world economy;
— to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and
— to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Member countries of the OECD are Austria, Belgium,Canada, Denmark, France, Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries became Members subsequently through accession at the dates indicated hereafter: Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971) and New Zealand (29th May 1973). The Commission of the European Communities takes part in the work of the OECD (Article 13 of the OECD Convention).

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full Member. NEA membership today consists of all European Member countries of OECD as well as Australia, Canada, Japan and the United States. The Commission of the European Communities takes part in the work of the Agency.

The primary objective of NEA is to promote co-operation among the governments of its participating countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source.

This is achieved by:

— *encouraging harmonization of national regulatory policies and practices, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;*
— *assessing the contribution of nuclear power to the overall energy supply by keeping under review the technical and economic aspects of nuclear power growth and forecasting demand and supply for the different phases of the nuclear fuel cycle;*
— *developing exchanges of scientific and technical information particularly through participation in common services;*
— *setting up international research and development programmes and joint undertakings.*

In these and related tasks, NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.

PREFACE

This document of the Nuclear Standards Subcommittee of the OECD/Nuclear Energy Agency Nuclear Data Committee (NEANDC) contains the 1991 version of the Nuclear Standards File, and summarizes the status of the individual nuclear standards as of the 29th meeting of the NEANDC in October 1991, with selective updating to May 1992.

The NEANDC was terminated in November 1991. The responsibility for the Nuclear Standards File was subsequently taken over by the new committee on Nuclear Science at OECD/NEA.

The Standards Subcommittee of the International Atomic Energy Agency Nuclear Data Committee (INDC) has collaborated through the exchange of technical information on those items which are common to the Standards Files of both Committees. The INDC Nuclear Standards File was published as an IAEA Technical Report No 227 in 1983. The present report is an update of the common items with that publication.

The objective of the file is to provide concise and readily usable reference guidelines to essential nuclear standards quantities for a variety of basic and applied endeavours.

The file consists of status summaries for eighteen nuclear data standards and data tabulations. The narrative summaries describe the current status of each of the standards and include references to recent relevant work and areas of continuing uncertainties. These brief reviews were prepared under the auspicies of the NEANDC and INDC by outstanding specialists in the respective fields.

The large majority of the recommended numerical data for the standard cross sections is taken from ENDF/B-VI, produced by the United States Cross Section Evaluation Working Group. The remainder of the numerical data is from evaluations undertaken by individuals or groups closely connected with the nuclear data activities promoted by the NEANDC and INDC. Generally, the numerical data tables include quantitative definitions of the data uncertainties and some guidelines as to their appropriate usage.

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Introduction

The majority of basic and applied nuclear data measurements are made relative to reference standards. It is essential that these standards are well defined, clearly referenceable and easily available. The INDC/NEANDC nuclear standards file provides such standard-reference quantities in a manner not otherwise available.

In order to improve the accuracy and consistency of experimental results it is recommended

> that standards tabulated in this report will be adopted for all measurements, and
>
> that when converting relative measured values to cross section values the numerical values given herein will be employed.

These recommendations will facilitate future evaluation work and ease later renormalizations when improved standard-reference information becomes available.

The standards file consists of tabulated reference values and a status summary for eighteen nuclear data standards.

The narrative summaries consist of concise and up-to-date statements delineating nuclear reference standards judged of importance by the Committee. These statements, prepared by selected specialists, outline the contemporary status (including shortcomings) and suggest possible avenues toward improvement. The statements explicitly support the accompanying numerical tabulations and set forth other important nuclear standards not amenable to straightforward numerical tabulation.

NEANDC STANDARDS SUBCOMMITTEE MEMBERSHIP
1991

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S Cierjacks	Germany
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INDC STANDARDS SUBCOMMITTEE MEMBERSHIP
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NEANDC/INDC Reference-Data-Type and Review Responsibilities

The responsibility for the individual reference data standards is shared among members of the NEANDC and INDC Standards Subcommittees and their delegates. The responsibility distribution as of 1991/1992 is as follows.

	Responsibilty	
Standard	National	Current Personnel
H(n,n)H	USA	G Hale/P Young
^{6}Li(n,t)^{4}He	USA	P Young/G Hale
^{10}B(n,α)^{7}Li	CBNM	E Wattecamps
C(n,n)C	USA	Y Fu/P Young
^{197}Au(n,γ)^{198}Au	CBNM	F Corvi
^{235}U(n,f)	UK/USSR	M Sowerby/V Konshin
^{235}U Fiss Fragm Anisotropy	CBNM	F J Hambsch
^{238}U(n,f)	Japan	Y Nakajima/Y Kanda
^{27}Al(n,α)	Austria	H Vonach
^{59}Co(n,2n)^{58}Co	Austria	H Vonach
^{93}Nb(n,2n)^{92}Nb	Austria	H Vonach
Neutron Energy Standards	Italy	C Coceva
Actinide Half-lives	CBNM/IAEA	W Bambynek/H Lemmel
Thermal parameters	France	H Tellier
Low Energy Cross Section Dependence	Belgium	C. Wagemans
^{252}Cf Fission Spectrum	Germany/IAEA Russia	W Mannhart/H Lemmel M Blinov
^{252}Cf nu-bar	Australia	J W Boldeman
Neutron Flux Comparison	France	E Fort/G Grenier
Gamma-ray Standards	France/IAEA	J Legrand/H Lemmel

THE H(n,n)H CROSS SECTION

This cross section is used as a standard neutron scattering cross section relative to which other elastic cross sections are measured in the MeV region. It is also the cross section for neutron flux measurements above about 0.5 MeV and is used for this purpose in several ways which together require a knowledge of the angular distribution in both hemispheres. Detection of proton recoils from hydrogenous radiators involves the cross section at backward angles, while a common method of measuring the relative response of organic scintillators to neutron energy is to scatter an incident monoenergetic neutron beam from hydrogenous samples.

In the case of organic scintillators frequent use is also made of computer codes for calculating the neutron detection efficiency for different thresholds as a function of energy, and in these calculations the differential scattering cross section is needed as input data.

THE H(n,n)H CROSS SECTION BELOW 20 MEV

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Status

The ENDF/B-VI cross sections for hydrogen represent the first new evaluation work on n-p scattering since those based on the Hopkins-Breit phase shifts were placed in the file. The new cross sections result from a charge-independent R-matrix analysis of n-p and p-p scattering at energies below 30 MeV that was done by Dodder and Hale (1). A summary of the channel configuration and data fitting characteristics of the analysis is given in Table I.

The R-matrix analysis includes many n-p measurements that were not available at the time of the Hopkins-Breit phase-shift analysis, and gives a representation of the n-p and p-p data in the 0-30 MeV range that is comparable to or better than that of other recent work (2,3). The new analysis also gives predictions for newly-measured observables, such as the polarization-transfer data from Karlsruhe (4), that look quite reasonable.

Table I. 0-30 MeV N-N R-Matrix Analysis

Channel	l_{max}	a_c (fm)
n-p	3	3.26
p-p	3	3.26

Reaction	# Observable types	# Data points	χ^2
n-p scattering	3	448	407
p-p scattering	4	388	399
Totals	7	836	806

parameters = 33 => χ^2 per degree of freedom = 1.004*

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* Including recent corrections (5) to the 16.9 MeV n-p analyzing-power data of Tornow et al (6) reduces the overall chi-square per degree of freedom of the fit to 0.9988.

The charge-independent model used takes the isospin-1 reduced-width amplitudes in the R-matrix describing n-p scattering to be identically the same as those describing p-p scattering. The energy eigenvalues in the two systems are taken to differ only by an overall constant Coulomb energy shift. This simple model allows the p-p scattering data to influence the n-p fit. We see in Fig 1, where measurements of the cross section and analyzing powers for the two reactions are compared, that the data are quite different at the same energy. These differences, coming primarily from Coulomb terms and symmetrization properties of the two systems, are well reproduced by the charge-independent calculation. The calculation also accounts well for the shape of the n-p angular distribution measurement (7) at 14 MeV.

Two quantities often used to characterize the center-of-mass n-p angular distribution near 14 MeV are the back-angle cross section, $\sigma(180°)$, and the asymmetry ratio

$$R = \sigma(180°) \ / \ \sigma(90°)$$

The ENDF/B-VI evaluation gives for these quantities at E_n = 14 MeV

$$\sigma(180°) = 58.89 \pm 0.60 \text{ mb}, \qquad R = 1.093 \pm 0.010$$

The R-value is in agreement with most previous measurements, but disagrees with a recent measurement of Ryves and Kolkowski (8) ($R = 1.053 \pm 0.015$) that is consistent with the ENDF/B-V value. The ENDF/B-VI values of the back-angle cross section and asymmetry ratio, on the other hand, are in excellent agreement with an evaluation of the 14.1 MeV data that was done in 1982 by Vincour, Bém and Presperin (9).

The fit to the n-p total cross section is compared to various measurements in Fig 2. These data, as well as accurate spin-dependent coherent cross-section measurements at thermal energies are well represented by the R-matrix calculation.

Values of the ENDF-B/VI standard total cross section and its provisionally assigned error is given at energies up to 20 MeV at the end of this section. Legendre coefficients for the center-of-mass differential cross section are also listed in this energy range, but their errors are not yet been determined. The total cross sections are quite similar to the previous Hopkins-Breit values. The angular distributions are somewhat more backward-peaked than the previous evaluation at energies near 14 MeV, corresponding to the relatively precise shape measurement shown in Fig 1.

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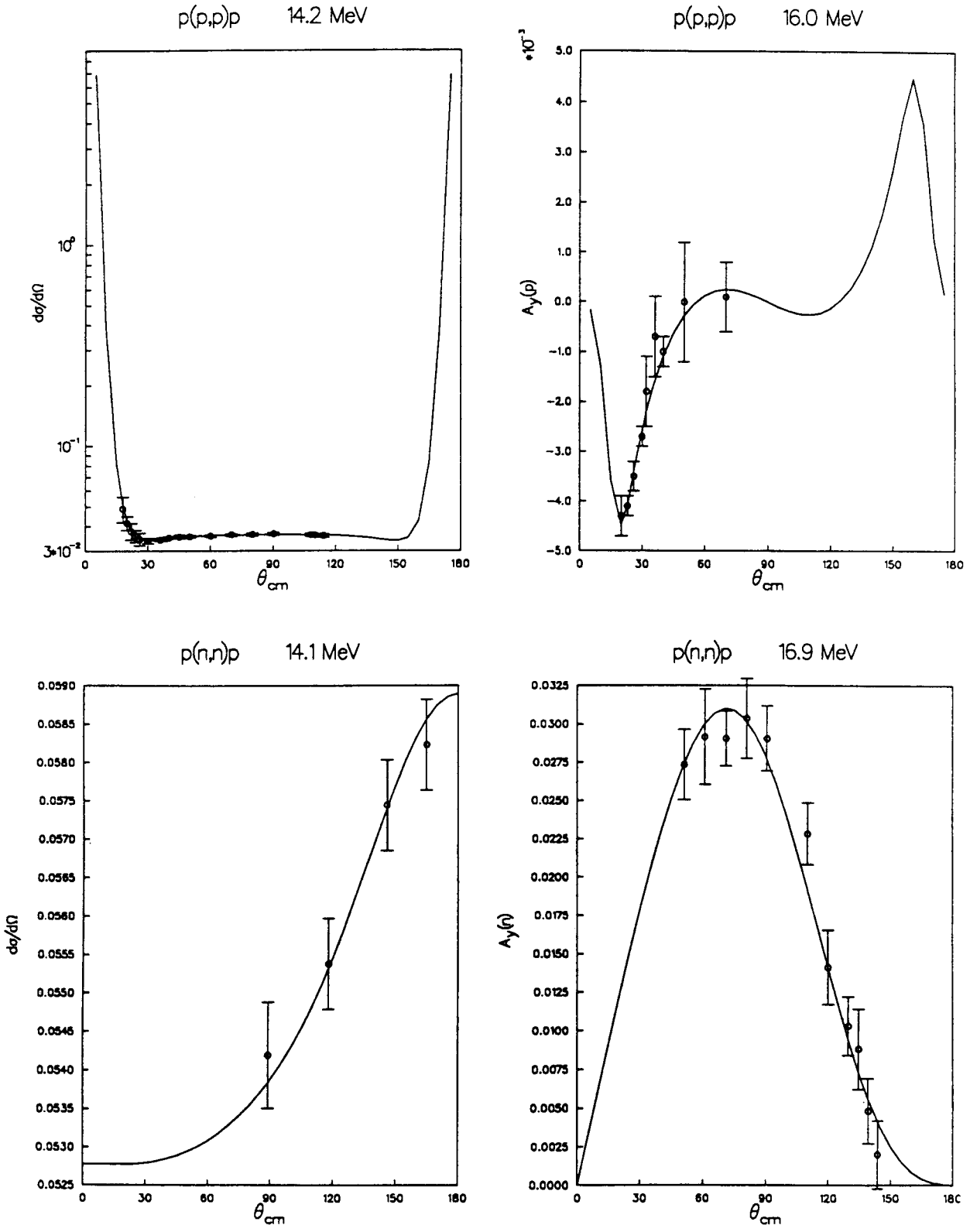

Fig. 1 Comparison of calculated and measured values of differential cross sections ($E_{n,p} \cong 14$ MeV) and analyzing powers ($E_{n,p} \cong 16.5$ MeV) for neutrons and protons incident on ^1H.

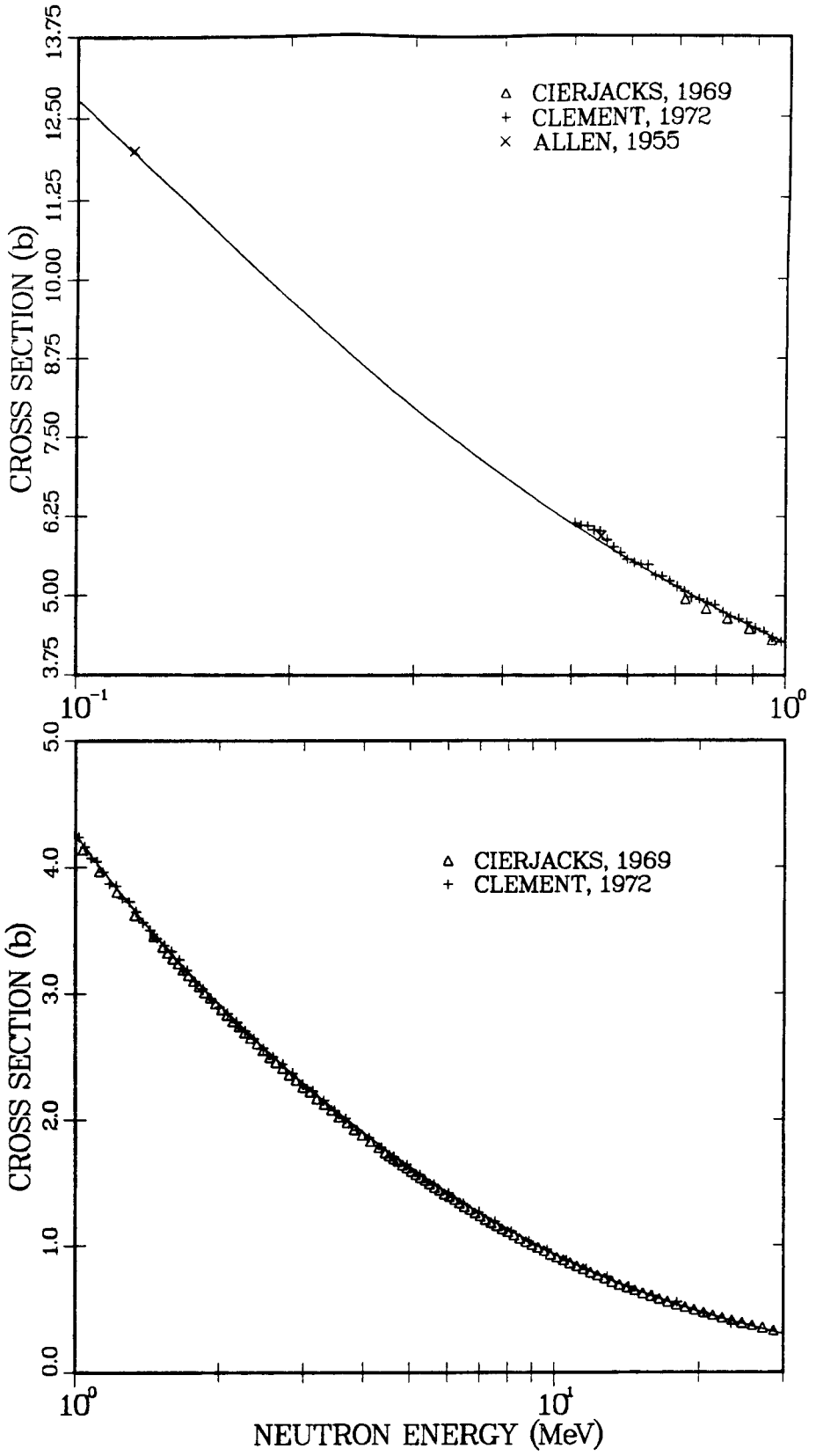

Fig. 2 Neutron total cross section for ^{1}H from R-matrix fit that resulted in the ENDF/B-VI standard $^{1}H(n,n)^{1}H$ cross section.

H(n,n) CROSS SECTIONS - Recommended Reference Data

Numerical values from ENDF/B-VI, MAT - 125
Applicable energy range 0.001 to 20.0 MeV
Log-log interpolation

Cross section values

E (keV)	XSEC (b)	E(keV)	XSEC(b)	E(keV)	XSEC(b)
1.00E00	2.0346E01	2.00E00	2.0121E01	3.00E00	1.9991E01
4.00E00	1.9899E01	5.00E00	1.9828E01	6.00E00	1.9667E01
8.00E00	1.9415E01	1.00E01	1.9223E01	1.50E01	1.8576E01
2.00E01	1.8130E01	2.50E01	1.7597E01	3.00E01	1.7173E01
3.50E01	1.6714E01	4.00E01	1.6325E01	4.50E01	1.5923E01
5.00E01	1.5571E01	5.50E01	1.5214E01	6.00E01	1.4894E01
6.50E01	1.4574E01	7.00E01	1.4283E01	7.50E01	1.3995E01
8.00E01	1.3730E01	8.50E01	1.3468E01	9.00E01	1.3226E01
9.50E01	1.2988E01	1.00E02	1.2765E01	1.10E02	1.2333E01
1.20E02	1.1951E01	1.30E02	1.1584E01	1.40E02	1.1255E01
1.50E02	1.0939E01	1.60E02	1.0652E01	1.70E02	1,0377E01
1.80E02	1.0125E01	1.90E02	9.8842E00	2.00E02	9.6610E00
2.20E02	9.2479E00	2.40E02	8.8781E00	2.60E02	8.5448E00
2.80E02	8.2429E00	3.00E02	7.9678E00	3.20E02	7.7162E00
3.40E02	7.4850E00	3.60E02	7.2718E00	3.80E02	7.0745E00
4.00E02	6.8912E00	4.20E02	6.7205E00	4.40E02	6.5611E00
4.60E02	6.4117E00	4.80E02	6.2715E00	5.00E02	6.1396E00
5.50E02	5.8412E00	6.00E02	5.5805E00	6.50E02	5.3504E00
7.00E02	5.1453E00	7.50E02	4.9611E00	8.00E02	4.7945E00
8.50E02	4.6429E00	9.00E02	4.5042E00	9.50E02	4.3765E00
1.00E03	4.2586E00	1.10E03	4.0471E00	1.20E03	3.8625E00
1.30E03	3.6992E00	1.40E03	3.5533E00	1.50E03	3.4219E00
1.60E03	3.3025E00	1.70E03	3.1934E00	1.80E03	3.0931E00
1.90E03	3.0003E00	2.00E03	2.9142E00	2.20E03	2.7588E00
2.40E03	2.6219E00	2.60E03	2.4999E00	2.80E03	2.3902E00
3.00E03	2.2907E00	3.20E03	2.1999E00	3.40E03	2.1166E00
3.60E03	2.0398E00	3.80E03	1.9686E00	4.00E03	1.9024E00
4.20E03	1.8406E00	4.40E03	1.7828E00	4.60E03	1.7286E00
4.80E03	1.6775E00	5.00E03	1.6294E00	5.20E03	1.5836E00
5.40E03	1.5407E00	5.60E03	1.4999E00	5.80E03	1.4610E00
6.00E03	1.4244E00	6.20E03	1.3891E00	6.40E03	1.3557E00
6.60E03	1.3237E00	6.80E03	1.2930E00	7.00E03	1.2639E00
7.50E03	1.1959E00	8.00E03	1.1345E00	8.50E03	1.0786E00
9.00E03	1.0277E00	9.50E03	9.8109E-01	1.00E04	9.3819E-01
1.05E04	8.9860E-01	1.10E04	8.6197E-01	1.15E04	8.2796E-01
1.20E04	7.9631E-01	1.25E04	7.6679E-01	1.30E04	7.3918E-01
1.35E04	7.1332E-01	1.40E04	6.8903E-01	1.45E04	6.6618E-01
1.50E04	6.4465E-01	1.55E04	6.2432E-01	1.60E04	6.0510E-01

E (keV)	XSEC (b)	E(keV)	XSEC(b)	E(keV)	XSEC(b)
1.65E04	5.8683E-01	1.70E04	5.6964E-01	1.75E04	5.5319E-01
1.80E04	5.3767E-01	1.85E04	5.2278E-01	1.90E04	5.0869E-01
1.95E04	4.9515E-01	2.00E04	4.8230E-01		

Uncertainties

Reviewers of the ENDF/B-VI standards cross sections which were evaluated by the US Cross Section and Evaluation Working Group (CSEWG) (see Introduction) have expressed the concern that the uncertainties resulting from the combination of R-matrix and simultaneous evaluations might have led to uncertainties that are too small. As a result, the CSEWG Standards Subcommittee at its May 1990 Meeting produced a set of expanded covariance estimates for the standard cross section reactions. These uncertainties are estimates such that if a modern day experiment were performed on a given standard cross section using the best techniques, approximately 2/3 of the results should fall within these expanded uncertainties.

Energy range (eV)	Uncertainty (Percent)
0.001 - 20 MeV	0.2

Relative Centre-of-Mass Neutron Angular Distributions

Legendre polynomial form: sum over A(I)*P(I), I = =,1,2,3 and 4, A(0) = 1, Linear-linear interpolation

E(keV)	A(1)	A(2)	A(3)	A(4)	A(5)	A(6)
1.00E00	-1.9193E-06	2.0948E-11	6.4660E-16	1.7526E-15	1.5960E-15	6.7523E-16
1.00E02	-1.7860E-04	2.8758E-07	-1.8308E-10	1.3878E-13	1.1367E-14	-6.1879E-15
2.00E02	-3.3629E-04	1.3983E-06	-1.2933E-09	3.1318E-12	2.0769E-14	-1.2353E-14
4.00E02	-6.1324E-04	7.0287E-06	-8.4343E-09	6.9385E-11	7.6969E-14	-2.2799E-14
6.00E02	-8.6197E-04	1.7978E-05	-2.4435E-08	4.2557E-10	3.9568E-13	-3.0769E-14
8.00E02	-1.0978E-03	3.4617E-05	-5.1971E-08	1.5353E-09	1.6840E-12	-1.7907E-14
1.00E03	-1.3289E-03	5.6997E-05	-9.4584E-08	4.1380E-09	5.4560E-12	7.2478E-14
2.00E03	-2.5208E-03	2.4908E-04	-7.5816E-07	8.7625E-08	2.1302E-10	1.1745E-11
4.00E03	-5.2452E-03	9.2990E-04	-9.4717E-06	1.7539E-06	7.1298E-09	1.0420E-09
6.00E03	-8.1589E-03	1.8023E-03	-4.5356E-05	9.6206E-06	4.3738E-08	1.4094E-08
8.00E03	-1.0913E-02	2.6839E-03	-1.3520E-04	3.0750E-05	1.0377E-07	8.7751E-08
1.00E04	-1.3277E-02	3.4647E-03	-3.0533E-04	7.2547E-05	-1.0052E-08	3.5407E-07
1.20E04	-1.5127E-02	4.0980E-03	-5.7383E-04	1.4049E-04	-9.7561E-07	1.0774E-06
1.40E04	-1.6412E-02	4.5940E-03	-9.4362E-04	2.3659E-04	-4.3368E-06	2.6764E-06
1.60E04	-1.7130E-02	5.0196E-03	-1.3966E-03	3.5926E-04	-1.2911E-05	5.6751E-06
1.80E04	-1.7304E-02	5.5057E-03	-1.8880E-03	5.0520E-04	-3.1091E-05	1.0523E-05
2.00E04	-1.6974E-02	6.2646E-03	-2.3394E-03	6.7379E-04	-6.4813E-05	1.7211E-05

THE H(n,n)H CROSS SECTION FROM 20 TO 350 MeV

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Status

VL40 is an energy-dependent partial-wave representation of combined (pp+np) elastic scattering data below 400 MeV. The np differential cross sections derived from VL40 are accepted by the NEANDC as a standard below 350 MeV; at higher energies it is felt that ignored inelasticity is no longer negligible. The solution was parameterized in each partial wave by the K-matrix form described in previously published VPI&SU analyses (1). Charge-splitting of isovector waves was accomplished by Coulomb barrier suppression and by separately parameterizing the low-energy $^1S_0$(np) state. The solution is summarized in Table I.

Table I. Summary of solution VL40

Channel	Data points	χ^2
pp(0-400 MeV)	1919	2792
np(0-400 MeV)	3026	4292

Lmax = 5, No of varied parameters = 54, OPEC by $g^2/4\pi = 13.5$

The cross section below 350 MeV can be characterized as being dominated by S-wave bound (or near-bound) states at threshold then falling very rapidly with energy to a shape which is dominated at forward and backward angles by peaking associated with one-pion-exchange (OPEC). This is illustrated in Figure 1 where the cross section is contour plotted from 0-180 degrees and from 20 to 360 MeV. The residuals shown in Figure 2 reveal the large abundance and distribution of cross section data (1606 points shown) in this kinematic regime. Most of the experiments, however, have large systematic errors so that any reasonable representation of the cross section must rely heavily upon other types of data (measurements involving polarized nucleons) and reasonable physical assumptions (such as isospin invariance)

VL40 cross sections can be compared to ENDF/B-VI standard cross sections (2) below 20 MeV by contour plotting the percentage difference in predicted cross sections from 0 to 180 degrees and from 10 to 30 MeV as in Figure 3. Below 20 MeV the agreement is quite good, generally within 1 %, but above 25 MeV (the extreme range of the ENDF analysis is at 30 MeV) the disagreement rises to around 10 % in a narrow forward region. Below 10 MeV the agreement is still very good, around 1 %, as the cross section rises to very high values.

Uncertainties are difficult to quantify since they are clearly dependent upon kinematics. We estimate, however, that they rise from a level of around 1 % at low energies (below 50 MeV) to a "few" percent at higher energies where the cross sections are much smaller and where the physics is much more involved. A study is currently underway to be more specific by comparing predicted cross sections from a variety of model "fits" (3).

Cross sections from VL40 can be obtained through the Scattering Analysis Interactive Dial-in system (SAID) at VPI&SU which is accessible through TELNET (VTINTE.PHYS.VT.EDU. or 128.173.7.3) with logon PHYSICS and Password QUANTUM. SAID also gives access to all of the measured data used in the analysis and to a number of other solutions including ENDF/B-VI. The cross sections are also encoded in a FORTRAN callable subroutine which is available upon request from VPI&SU.

This work was supported in part by the U.S. Department of Energy Grant DE-AS05-76ER04928.

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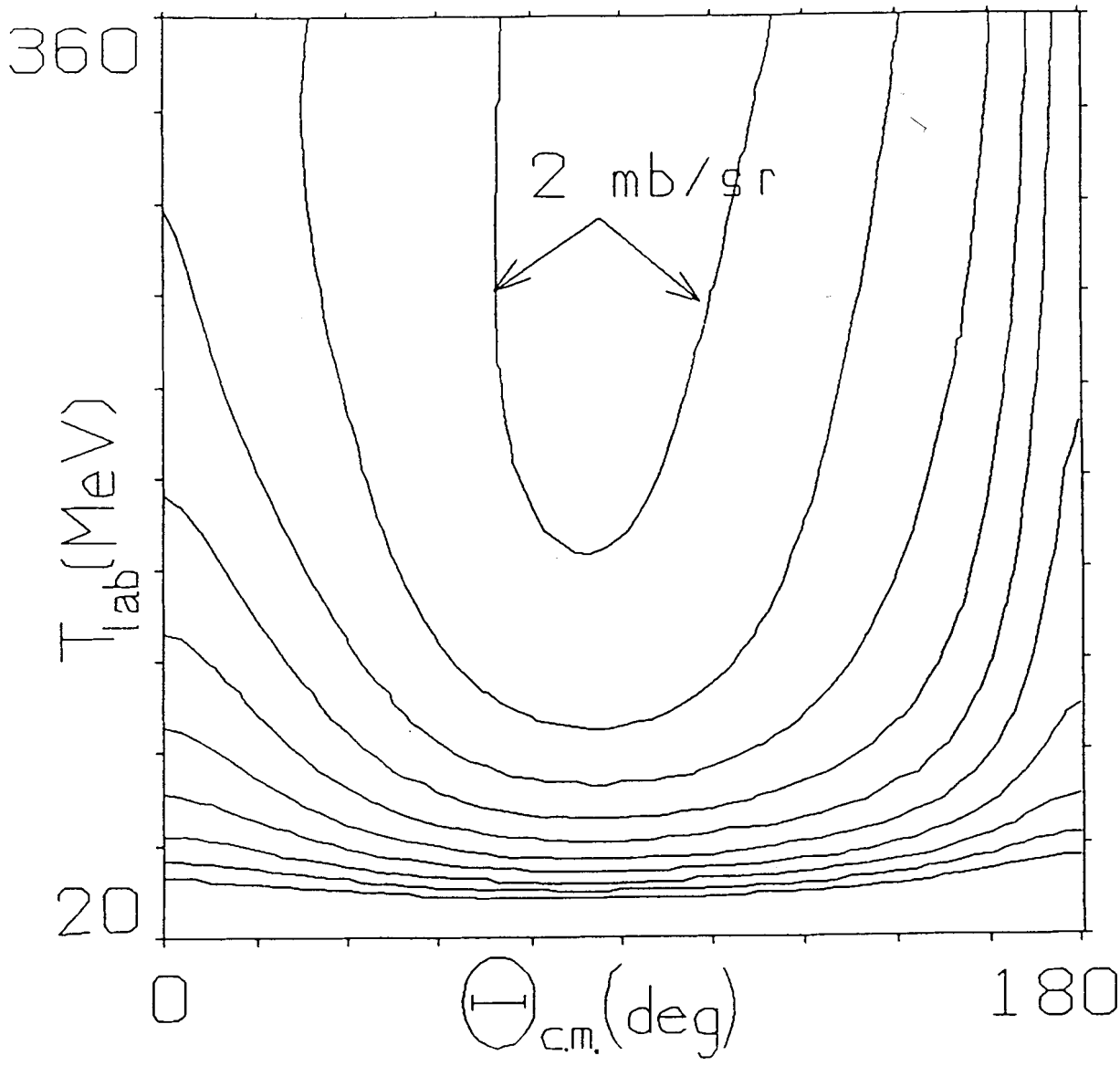

Figure 1- Contour plot of np differential cross section from 0-180 degrees, 20-360 MeV. Contours are from 2 mb/sr to 20 mb/sr, in increments of 2 mb/sr, with the cross section minimum (around 1.4 mb/sr) at 90 degrees and 360 MeV.

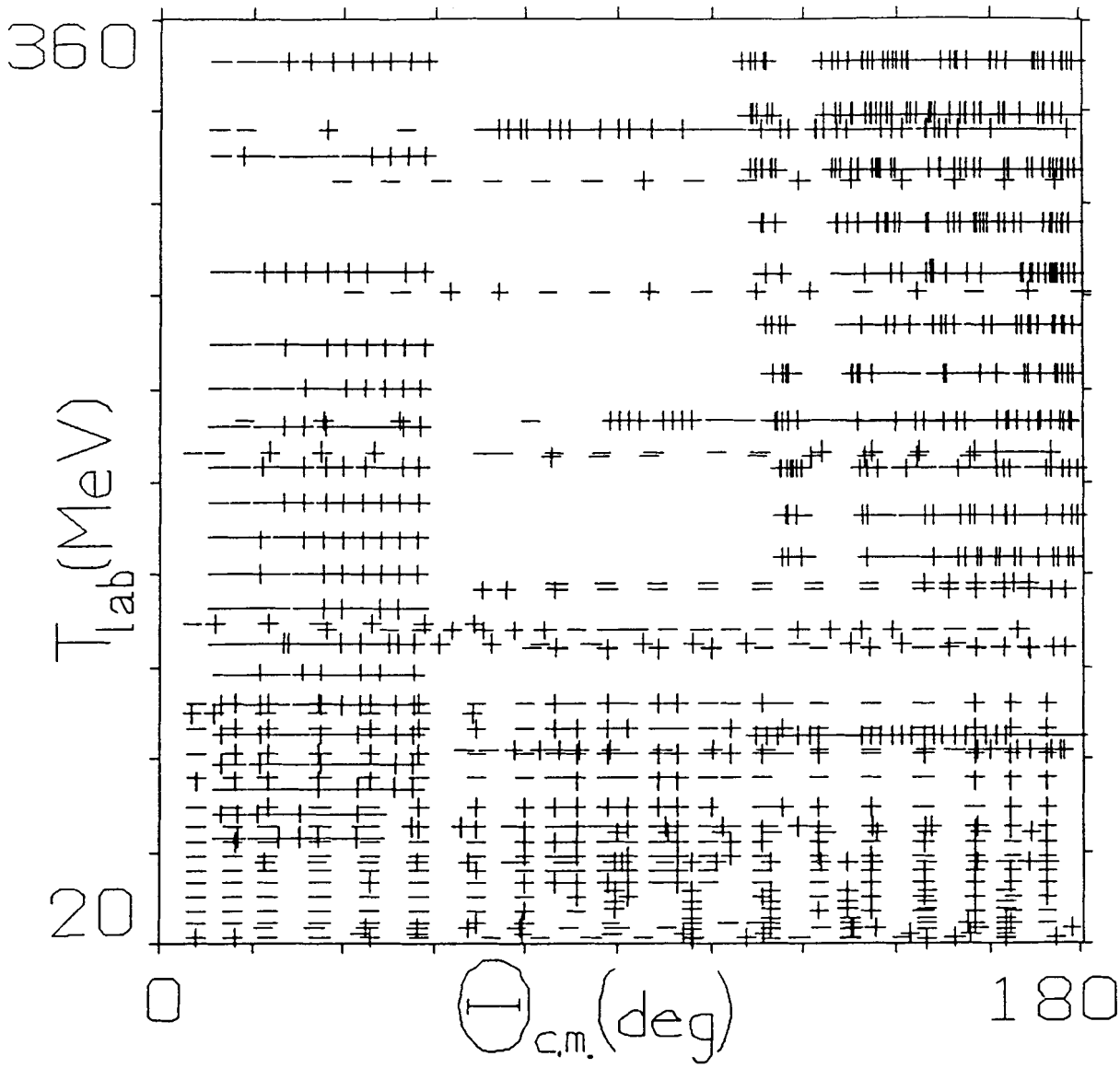

Figure 2- Density and distribution of measured data. The +/- symbols are data-base residuals ([theory-experiment]/error).

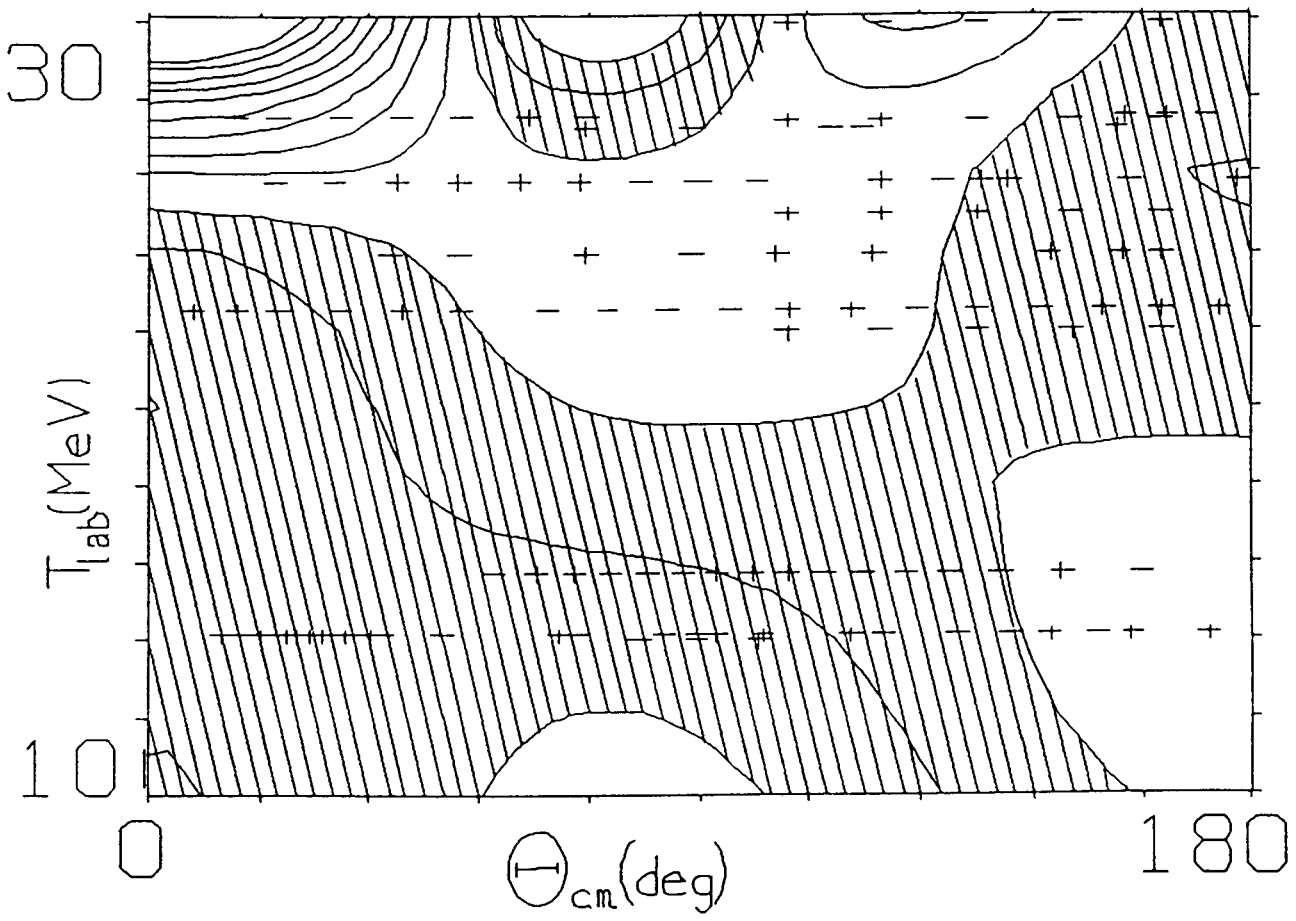

Figure 3- Contour plot of the percentage difference between cross sections predicted by VL40 and ENDF/B-6. The contour levels are in 1% increments and the shaded area shows the domain over which agreement is within 1%. The +/- symbols are as defined in Figure 2.

THE $^{6}Li(n,t)^{4}He$ CROSS SECTION

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Because of its relatively large cross section and positive Q-value, and the convenience of counting the light triton and alpha-particle products, this reaction is widely used as a standard. The recommended energy range for use as a standard is thermal to 100 keV. At ~100 keV the cross section begins to deviate substantially from 1/v behavior. However, applications in which the cross section is used as a standard at energies above the 240 keV resonance and up to a few MeV are not uncommon. The cross section at energies up to several MeV is also of interest because lithium is envisaged as a tritium-breeding medium in most fusion designs. With the standards application mainly in mind, we will limit the discussion of this review to energies below 1 MeV.

Analysis Summary of the Evaluation of the $^{6}Li(n,t)^{4}He$ Standard Cross Section for ENDF/B-VI

The evaluation of the $^{6}Li(n,t)^{4}He$ standard cross section was included in the comprehensive standards analysis that was provided for ENDF/B-VI (1,2). In this analysis part of the experimental data base was included in a simultaneous least-squares analysis and part in a comprehensive R-matrix analysis. Results from the two independent analyses were then combined in a third step, taking full account of the covariance matrices that resulted from the separate analyses.

It was felt by the Standards Subcommittee of the US Cross Section Evaluation Working Group that a simultaneous analysis was necessary for ENDF/B-VI in order to properly take into account all the cross section data available for standards reactions. In addition to the $^{6}Li(n,t)^{4}He$ cross section data the simultaneous analysis included absolute cross sections and cross section ratios for the $^{6}Li(n,n)^{6}Li$, $^{10}B(n,\alpha_0)^{7}Li$, $^{10}B(n,\alpha_1)^{7}Li^*$, $^{10}B(n,n)^{10}B$, $^{197}Au(n,\gamma)^{198}Au$, $^{235}U(n,f)$, $^{238}U(n,\gamma)^{239}U$, $^{238}U(n,f)$ and $^{239}Pu(n,f)$ reactions.

Because the $^{6}Li(n,t)^{4}He$ and $^{10}B(n,\alpha_0)^{7}Li$ reactions are dominated by resonances in the standards region, the CSEWG Standards Subcommittee felt it was equally important that R-matrix analyses that accurately specify the energy dependence of the reactions also be factored into the standards analysis,.The R-matrix analysis for n + ^{6}Li included experimental data for the n + ^{6}Li total cross section, $^{6}Li(n,n)^{6}Li$ and $^{4}He(t,t)^{4}He$ elastic differential cross sections, as well as the $^{6}Li(n,t)^{4}He$ standard cross section. The most important ^{6}Li total cross section data included in the R-matrix analysis were the data of Harvey and Hill (3) and Smith et al (4). The $^{6}Li(n,n)^{6}Li$ elastic scattering measurements used in the analysis were from the work of Smith et al (5) as

well as the differential cross sections measured by Lane et al (6). Finally, the $^6Li(n,t)^4He$ standard cross section data included in the analysis were from the work of Renner et al (7), Lamaze et al (8), Brown et al (9), Overley et al (10) and Bartle (11).

Results

The n + ^6Li total cross section between E_n = 0.1 keV and 2 MeV that resulted from the ENDF/B-VI analysis is compared in Figure 1 with the previous ENDF/B-V evaluation and with the experimental data of Harvey and Hill (3). Similarly, the analyzed $^6Li(n,t)^4He$ standard cross section between 10 keV and 2 MeV is compared with ENDF/B-V and with the measurements of Renner et al (7), Lamaze et al (8) and Bartle (11) in Figure 2. Note that although the energy range shown in Figures 1 and 2 extends to 2 MeV, the comprehensive standards analysis resulted in cross sections only to 1 MeV.

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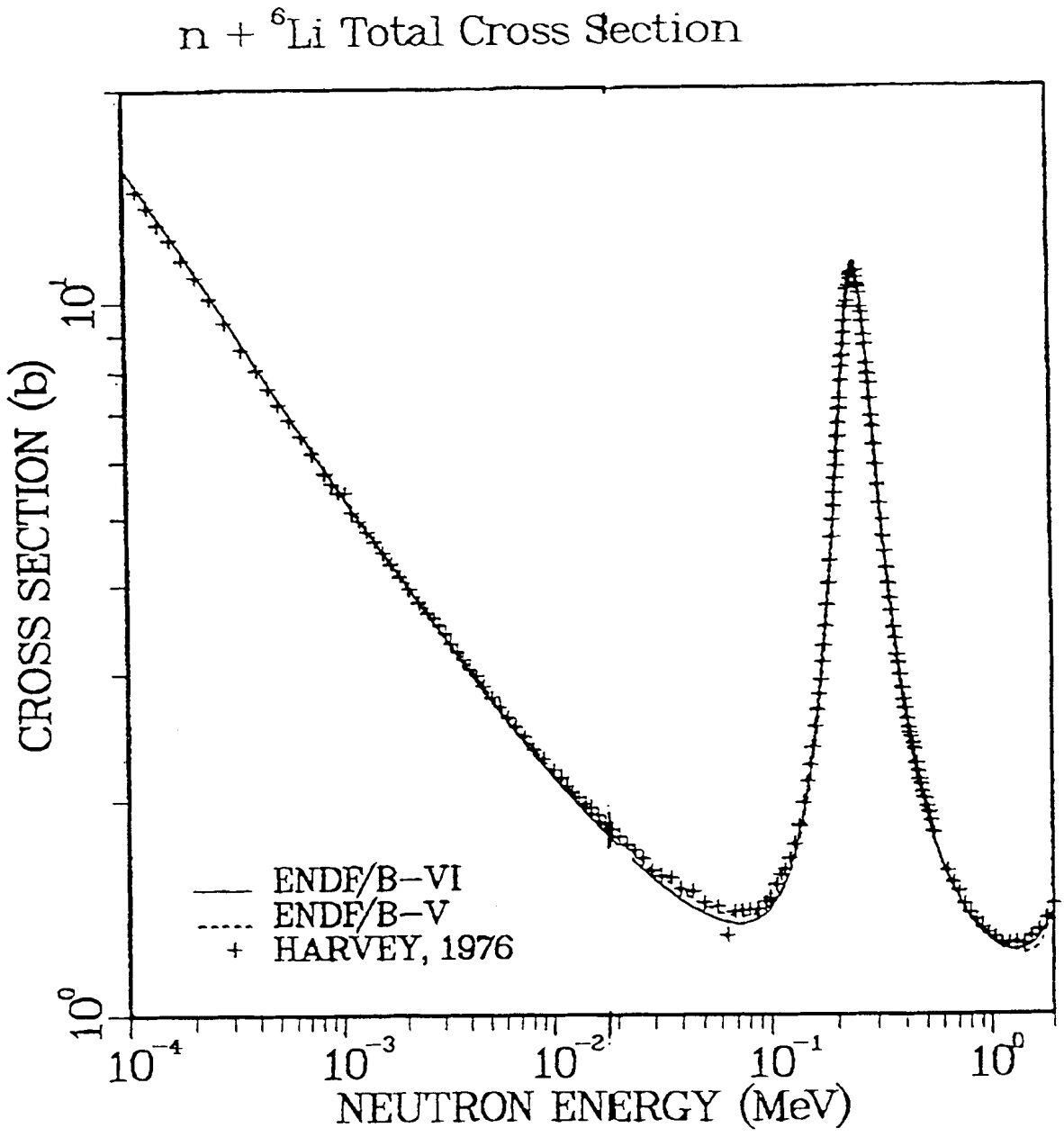

Figure 1 The total cross section for the n + ^{6}Li interaction between 0.1 keV and 2 MeV. The solid curve is ENDF/B-VI; the dashed curve is ENDF/B-V.2; and the experimental data are from Reference 3.

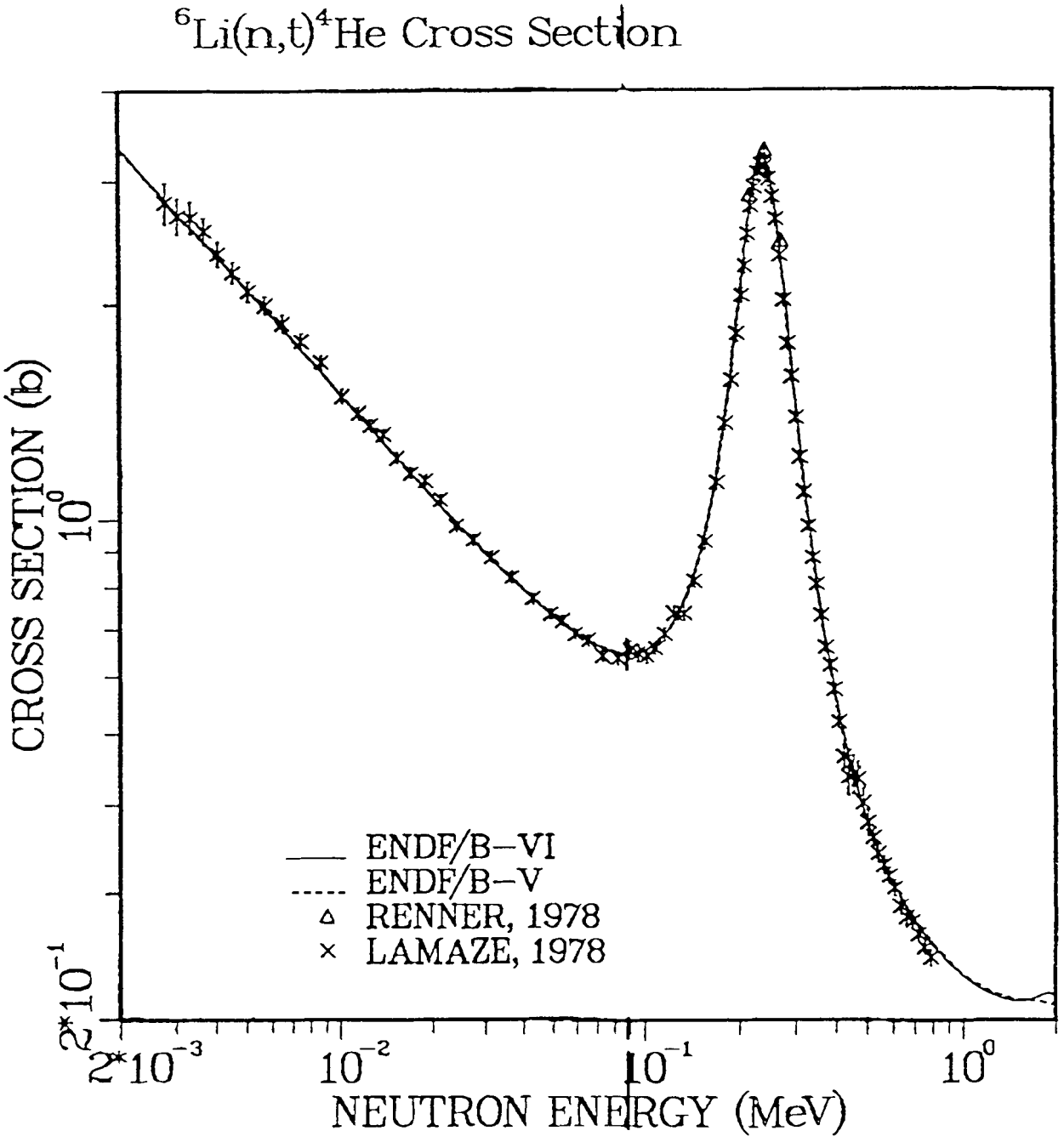

Figure 2 The ^{6}Li(n,t)^{4}He standard cross section between 10 keV and 2 MeV. The solid curve is ENDF/B-VI; the dashed curve is ENDF/B-V.2; and the experimental data are from References 7 and 8.

^{6}Li(n,t) CROSS SECTIONS - Recommended Reference Data

Numerical values from ENDF/B-VI, MAT 325
Applicable energy range thermal - 100 keV
Log-Log interpolation

Cross Section Values

E (keV)	XSEC(b)	E(keV)	XSEC(b)	E(keV)	XSEC(b)
1.00E-08	4.7342E04	2.53E-05	9.4098E02	9.40E-03	4.8793E01
1.50E-01	1.2196E01	2.50E-01	9.4428E00	3.50E-01	7.9777E00
4.50E-01	7.0337E00	5.50E-01	6.3613E00	6.50E-01	5.8502E00
7.50E-01	5.4454E00	8.50E-01	5.1137E00	9.50E-01	4.8371E00
1.50E00	3.8470E00	2.50E00	2.9791E00	3.50E00	2.5181E00
4.50E00	2.2214E00	5.50E00	2.0110E00	6.50E00	1.8516E00
7.50E00	1.7243E00	8.50E00	1.6221E00	9.50E00	1.5358E00
1.50E01	1.2301E00	2.00E01	1.0737E00	2.40E01	9.8670E-01
3.00E01	8.9300E-01	3.50E01	8.3590E-01	4.50E01	7.5620E-01
5.50E01	7.0520E-01	6.50E01	6.7250E-01	7.50E01	6.5320E-01
8.50E01	6.4500E-01	9.50E01	6.4680E-01	1.00E02	6.5150E-01

Uncertainties

Reviewers of the ENDF/B-VI standards cross sections which were evaluated by the US Cross Section Evaluation Working Group (CSEWG) (see Introduction) have expressed the concern that the uncertainties resulting from the combination of R-matrix and simultaneous evaluations might be too small. As a result, the CSEWG Standards Subcommittee at its May 1990 Meeting produced a set of expanded covariance estimates for the standard cross section reactions. These uncertainties are estimates such that if a modern day experiment were performed on a given standard cross section using the best techniques, approximately 2/3 of the results should fall within these expanded uncertainties. The expanded uncertainties for the ^{6}Li(n,t) cross section are given in the following table and are compared to values from the combined output of the standards covariance analysis.

Energy Range (keV)	Expanded Uncertainty (percent)	Combined Analysis (percent)
1.00E-08 - 0.1	0.3	0.14
0.1 - 1.0	0.5	
1.0 - 10.0	0.7	0.14
10.0 - 50.0	0.9	
50.0 - 90.0	1.1	0.25
90.0 - 150.0	1.5	
150.0 - 450.0	2.0	0.29

THE $^{10}B(n,\alpha)^7Li$ CROSS SECTION

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Natural boron or ^{10}B-enriched samples are often used for neutron induced reaction rate measurements or for neutron flux determination. A large variety of detectors is available and the reaction underlying the detection system is either $^{10}B(n,\alpha\gamma)^7Li$ or $^{10}B(n,\alpha)^7Li$.

Different symbols are used. The α_0 refers to an α-particle emission with energy E_a=1.7891 MeV, leaving the residual nucleus ^7Li in its ground state. The α_1 refers to an α-emission with energy of E_α=1.4832 MeV, leaving the residual nucleus ^7Li in its first excited state, which decays by prompt emission of a gamma-ray of 478.5 keV, with isotropic angular distribution in the center of mass system. The symbol $(n,\alpha\gamma)$ is identical with (n,α_1), whereas (n,α), also called total (n,α), is equal to the sum of (n,α_0) and (n,α_1). The branching ratio is defined as $R = \sigma(n,\alpha_1)/(\sigma(n,\alpha_0)+\sigma(n,\alpha_1))$. In the ENDF/B-VI standard file the $^{10}B(n,\alpha_0)$ and the $^{10}B(n,\alpha_1)$ cross sections are considered as standard reference data in the energy range from thermal energy to 200 keV.

Status of the requests

In WRENDA 87/88 there are eight requests pending for $^{10}B(n,\alpha)$ absolute cross section measurements, and three requests for branching ratio measurements as well, with accuracies ranging from 3% down to as low as 1%. There is one request for total cross section data from 1 keV to 2 MeV, with 1 % accuracy, and one request for capture cross section measurements from 25 meV to 200 keV with 20 % accuracy.

Requests for cross section measurements of $^{10}B(n,\alpha)$, $^{10}B(n,\alpha_1)$ and for the branching ratio, are also made in the high priority request list refs (1) and (2). A high priority request for $^{10}B(n,\alpha_0)$ measurements was submitted, by A D Carlson, May 92, to the U.S. compilation of requests for Nuclear Data.

As mentioned in the progress report of June 92 of the NEA-NSC International Interlaboratory Collaboration on the $^{10}B(n,\alpha)$ standard cross section, (3), there is a need to extend the energy range over which the cross section can be used as a standard so that a smooth and well overlapping transition to higher energy standards such as H(n,n) can be achieved.

Status of the recommended reference data

A review of measured cross section data for the neutron interactions with ^{10}B was made by A D Carlson (4) in 1983, and also by W P Poenitz (5) in 1984 and it was shown that the data base was poor.

At the IAEA Conference on Nuclear Standard Reference Data, held in Geel, November 1984, W P Poenitz presented a paper on : "The simultaneous evaluation of interrelated cross sections by generalized least squares and related data file requirements" (6). The cross sections involved are: $^{6}Li(n,\alpha)$, $^{6}Li(n,n)$, $^{10}B(n,\alpha_0)$, $^{10}B(n,\alpha_1)$, $^{10}B(n,n)$ $^{197}Au(n,\gamma)$, $^{238}U(n,\gamma)$ $^{235}U(n,f)$, $^{238}U(n,f)$ and $^{239}Pu(n,f)$. The experimental data included are:

* Absolute measurements of: cross sections, ratio of two cross sections, sums of cross sections, or ratio of a cross section vs. the sum of cross sections.
* Relative shape measurements of: cross sections, ratios of two cross sections, sums of cross sections, or ratio of a cross section vs. the sum of cross sections.
* The integral of a cross section over a (fission) neutron spectrum.

Uncertainties for each experiment, correlations in that experiment, as well as correlations with other experiments are part of the input, and a full covariance analysis is performed.

In 1985 A.D. Carlson et al. (7) presented the initial phase in the development of the new ENDF/B-VI file. This standards evaluation is different from earlier ENDF/B evaluations and has three separate parts:

* Simultaneous evaluation using generalized least squares program (6)
* R-matrix evaluation, by G.M. Hale (8), relying on a large number of reaction channels of the ^{11}B system, including inverse reactions,
* Procedure for combining the simultaneous and the R-matrix evaluation (9).

Cross section values for ^{10}B $(n,\alpha\gamma)$ and ^{10}B (n,α) from the simultaneous evaluation are higher than the corresponding values from the R-matrix analysis above 50 keV, and these discrepancies are believed to be explained by the poor data base.

In October 1987 the ENDF/B-VI $^{10}B(n,\alpha_1)$ and $^{10}B(n,\alpha_0)$ standard cross section data were released together with surprisingly small uncertainties, called hereafter combined uncertainty. Numerical values are listed in Table 1 and 2 and also drawn in Figure 1.

The evaluation procedure underlying the ENDF/B-VI standards file is a very large and successful undertaking that profits of all available and related

information. It reduces the arbitrariness in evaluating the quality of a specific measurement and it provides a complete covariance matrix.

Reviewers of the ENDF/B-VI standards cross sections, which were evaluated by the US Cross Section and Evaluation Working Group (CSEWG), have expressed the concern that the uncertainties resulting from the combination of R-matrix and simultaneous evaluations might be too small. As a result, the CSEWG Standards Subcommittee at its 1990 Meeting produced a set of expanded covariance estimates for standard cross section reactions. These uncertainties are estimates such that if a modern day experiment were performed on a given standard cross section using the best techniques, approximately two-thirds of the results should fall within these expanded uncertainties. The expanded uncertainties for the $^{10}B(n,\alpha_0)$ and $^{10}B(n,\alpha_1)$ cross section are also given in Table 1, and we observe a substantial difference between combined and expanded uncertainties, thus emphasizing the need for more accurate measurements if one has to achieve the requested accuracy.

Status of recent and ongoing measurements

An Interlaboratory Collaboration was formed providing a mechanism to improve the $^{10}B(n,\alpha)$ cross sections, in particular in view of reducing the large difference between the combined and the expanded uncertainties and in view of increasing the high energy limit from 200 keV towards higher energies. From the progress report of the Interlaboratory Collaboration, June 92, (3), and from recent publications one can see that appreciable efforts are devoted in recent times to measure more accurately $^{10}B(n,\alpha)$ cross sections and related data.

* Recent branching ratio measurements from 100 to 600 keV, by L W Weston and J H Todd at ORNL, (10), are 20 to 25 % lower than the ENDF/B-VI values, thus indicating a problem that is due either to $^{10}B(n,\alpha_1)$ or $^{10}B(n,\alpha_0)$, or to both.

* New measurements of the $^{10}B(n,\alpha_1)$ cross section in the 0.3 to 4 MeV neutron energy interval were performed by R A Schrack et al at ORNL, (11), with flux determination by a "black" plastic scintillator. Below 1 MeV the agreement with ENDF/B-VI is excellent (except for a 5 % difference at 420 keV), but above 1.5 MeV major unexplained differences arise.

* Unpublished measurements, under analysis, of the $^{10}B(n,\alpha_1)$ cross section by R A Schrack et al in the 5 keV to 1 MeV neutron energy range with flux determination by a gaseous hydrogen filled proportional counter give lower values than ENDF/B-VI in the neutron energy range from 100 to 500 keV.

* Measurements of total n,α yield and angular distribution of α particle yield relative to $^{235}U(n,f)$ were performed by R C Haight at WNR and are in the analysis stage.

* Measurements of the total cross section by A Brusegan at CBNM are foreseen in Autumn 92, and the samples are available now. Up to 100 keV the background determination will rely on the black resonance transmission method and for higher energies (possibly up to 300 keV) on comparison of ^{6}Li and ^{7}Li time of flight distributions.

* Total cross section measurements for higher energies, but overlapping at the low energy side with the CBNM measurements, are scheduled in Autumn 92, by J A Harvey and O A Wasson at ORNL

Ongoing experiments will give rise to new data, and quite some work will remain to implement the new data in the simultaneous evaluation of the large number of interrelated data. It will be exciting to see the impact of single new measurements on the global fit.

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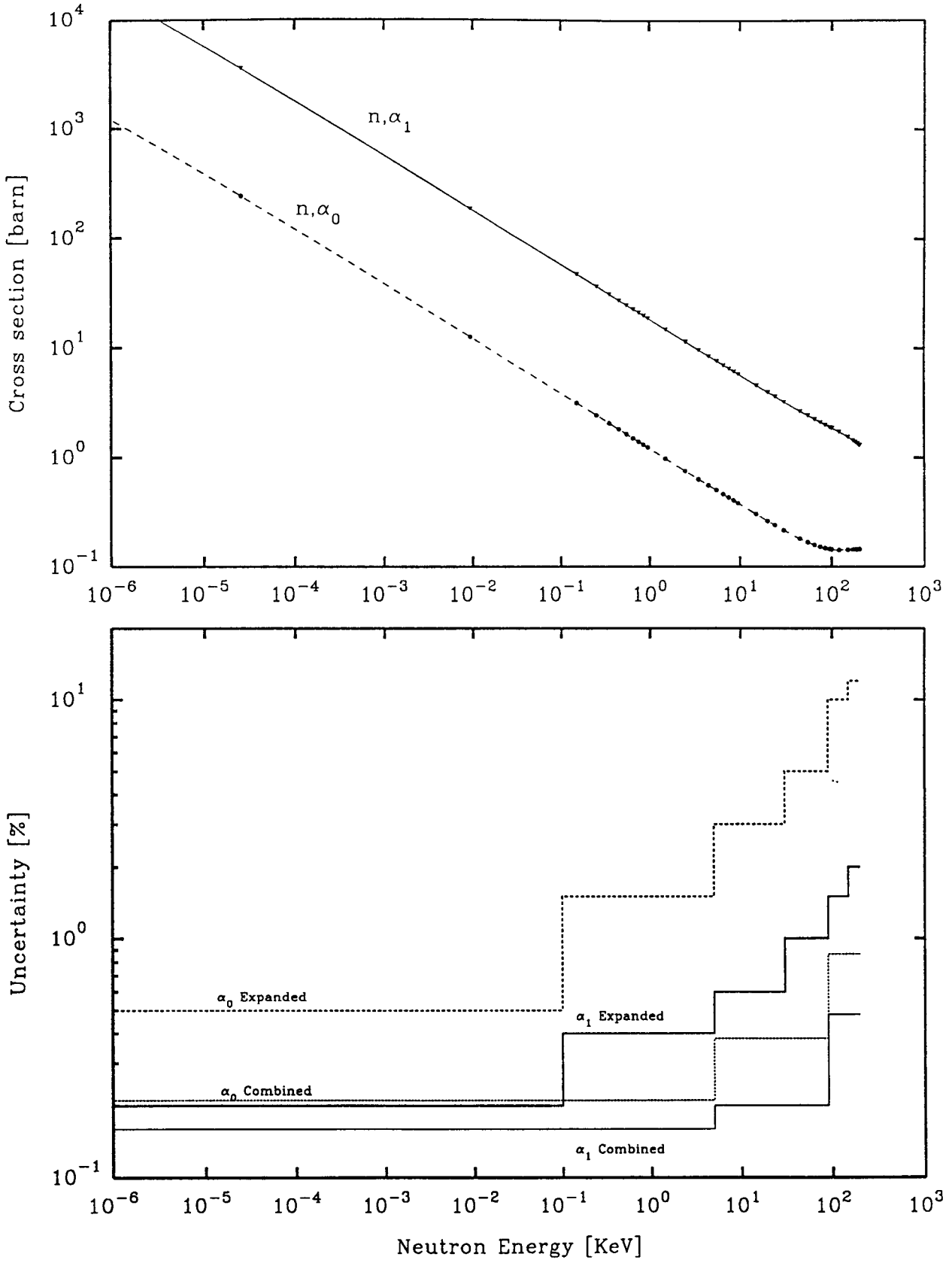

Fig.1. Results of the ENDF/B- VI standards evaluation (EVAL-JUL87, MAT 525)
- Top : cross-section values for the reactions (n,α_1) and (n,α_0)
- Bottom : the uncertainties "combined" and " expanded" for each cross section

$^{10}B(n,\alpha_0)$ CROSS SECTION - Recommended Reference Data

Table 1: Numerical values from ENDF/B-VI, MAT-525
Applicable energy range thermal to 200 keV
Log-log interpolation

Cross section values

E (keV)	XSEC(b)	E(keV)	XSEC(b)	E(keV)	XSEC(b)
1.00E-08	1.2140E04	2.53E-05	2.4127E02	9.40E-03	1.2500E01
1.50E-01	3.1169E00	2.50E-01	2.4105E00	3.50E-01	2.0347E00
4.50E-01	1.7928E00	5.50E-01	1.6199E00	6.50E-01	1.4889E00
7.50E-01	1.3850E00	8.50E-01	1.3000E00	9.50E-01	1.2290E00
1.50E00	9.7480E-01	2.50E00	7.5190E-01	3.50E00	6.3330E-01
4.50E00	5.5710E-01	5.50E00	5.0280E-01	6.50E00	4.6170E-01
7.50E00	4.2910E-01	8.50E00	4.0250E-01	9.50E00	3.8040E-01
1.50E01	3.0170E-01	2.00E01	2.6130E-01	2.40E01	2.3900E-01
3.00E01	2.1480E-01	4.50E01	1.7960E-01	5.50E01	1.6620E-01
6.50E01	1.5720E-01	7.50E01	1.5100E-01	8.50E01	1.4670E-01
9.50E01	1.4400E-01	1.00E02	1.4310E-01	1.20E02	1.4120E-01
1.50E02	1.4170E-01	1.70E02	1.4260E-01	1.80E02	1.4310E-01
1.90E02	1.4340E-01	2.00E02	1.4380E-01		

Uncertainties

Reviewers of the ENDF/B-VI standards cross sections which were evaluated by the US Cross Section and Evaluation Group (CSEWG) (see Introduction) have expressed the concern that the uncertainties resulting from the combination of R-matrix and simultaneous evaluations might be too small. As a result, the CSEWG Standards Subcommittee at its May 1990 Meeting produced a set of expanded covariance estimates for the standard cross section reactions. These uncertainties are estimates such that if a modern day experiment were performed on a given standard cross section using the best techniques, approximately 2/3 of the results should fall within these expanded uncertainties. The expanded uncertainties for the $^{10}B(n,\alpha_0)$ cross section are given in the following table and are compared to values from the combined output of the standards covariance analysis.

Energy range (keV)	Expanded uncertainty (percent)	Combined uncertainty (percent)
1.0E-08 - 0.1	0.5	0.21
0.1 - 5.0	1.5	
5.0 - 30.0	3.0	0.38
30.0 - 90.0	5.0	
90.0 - 150.0	10.0	0.86
150.0 - 200.0	12.0	
200.0 - 250.0	15.0	0.79

$^{10}B(n,\alpha_1)$ CROSS SECTIONS - Recommended reference data

Table 2: Numerical values from ENDF/B-VI, MAT-525
Applicable energy range thermal - 200 keV
Log-log interpolation

Cross section values

E(keV)	XSEC(b)	E(keV)	XSEC(b)	E(keV)	XSEC(b)
1.00E-08	1.8104E05	2.53E-05	3.5982E03	9.40E-03	1.8644E02
1.50E-01	4.6500E01	2.50E-01	3.5967E01	3.50E-01	3.0366E01
4.50E-01	2.6752E01	5.50E-01	2.4176E01	6.50E-01	2.2229E01
7.50E-01	2.0681E01	8.50E-01	1.9413E01	9.50E-01	1.8345E01
1.50E00	1.4560E01	2.50E00	1.1235E01	3.50E00	9.4717E00
4.50E00	8.3353E00	5.50E00	7.5254E00	6.50E00	6.9123E00
7.50E00	6.4260E00	8.50E00	6.0296E00	9.50E00	5.6981E00
1.50E01	4.5194E00	2.00E01	3.9090E00	2.40E01	3.5682E00
3.00E01	3.1959E00	4.50E01	2.6285E00	5.50E01	2.3943E00
6.50E01	2.2203E00	7.50E01	2.0844E00	8.50E01	1.9745E00
9.50E01	1.8835E00	1.00E02	1.8420E00	1.20E02	1.7031E00
1.50E02	1.5330E00	1.70E02	1.4303E00	1.80E02	1.3817E00
1.90E02	1.3340E00	2.00E02	1.2876E00		

Uncertainties

Reviewers of the ENDF/B-VI standards cross sections which were evaluated by the US Cross Section and Evaluation Group (CSEWG) (see Introduction) have expressed the concern that the uncertainties resulting from the combination of R-matrix and simultaneous evaluations might be too small. As a result, the CSEWG Standards Subcommittee at its May 1990 Meeting produced a set of expanded covariance estimates for the standard cross section reactions. These uncertainties are estimates such that if a modern day experiment were performed on a given standard cross section using the best techniques, approximately 2/3 of the results should fall within these expanded uncertainties. The expanded uncertainties for the $^{10}B(n,\alpha\gamma)$ cross section are given in the following table and are compared to values from the combined output of the standards covariance analysis.

Energy range (keV)	Expanded uncertainty (percent)	Combined uncertainty (percent)
1.0E-08 - 0.1	0.2	0.16
0.1 - 5.0	0.4	
5.0 - 30.0	0.6	0.20
30.0 - 90.0	1.0	
90.0 - 150.0	1.5	0.48
150.0 - 200,0	2.0	
200.0 -250.0	2.5	0.62

THE C(n,n) CROSS SECTION

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The primary use of this cross section is as a scattering standard at energies of less than 2 MeV where the neutron total and elastic scattering cross sections are essentially identical. The cross section in this energy range varies slowly with energy and is largely free of resonance structure. Similar use as a standard can be made to ~ 4.8 MeV (i.e. to the inelastic scattering threshold) if care is taken to avoid prominent resonance structure. The target mass is sufficiently heavy to reduce centre-of-mass energy loss to amounts well below that encountered in the use of the primary H(n,n) scattering standard. Prominent resonances, notably at 2.087 MeV, provide good energy reference points and, at some energies (e.g. near 3.5 MeV) the elastic scattering distributions display well-defined minima that are useful in experimental angle calibrations.

Status and Recent Results

The ENDF/B-V differential cross sections for neutron scattering from natural carbon below 2 MeV, recommended as standards for measurements and based on an R-matrix analysis for ^{12}C using natural carbon data (1), were revised for ENDF/B-VI to include ^{13}C resonances for high-resolution applications (2). The ^{13}C cross sections are also based on an R-matrix analysis of the available data. The 0.1529 and 1.736-MeV resonances rise above the natural background by 7 and 1 %, respectively.

The target spin 1/2 for ^{13}C required the use of a multichannel multilevel R-matrix analysis to interpret the available data and to generate angular distributions near the two resonances. The angular distributions needed have not yet been measured, but they can be reliably obtained from the R matrix analysis if the spins and parities of the resonances are well known, as appears to be the case for the two relevant resonances. The R-matrix analysis of Lane et al (3) for ^{13}C using total and differential data for ^{13}C from 1.25 to 6.5 MeV, provided a solid foundation for this purpose. Additional data at thermal energy (4) and near the 0.1529-MeV resonance (5) completed the required data base for the new ^{13}C evaluation.

Combination of the R-matrix results for ^{13}C with the ENDF/B-V evaluated data for natural carbon is not completely straightforward because the smoothed cross sections of ^{13}C, including the resonances, were already absorbed in the recommended results for natural carbon. Therefore, the cross sections of natural

carbon recommended previously in energy regions far from the ^{13}C resonances should not be changed and some compensation for the newly added resonance areas around each resonance should be made. With some trial and error, the following cross-section file was generated first:

0.00 to 0.10 MeV: previously evaluated natural carbon data
0.10 to 0.13 MeV: linear interpolation
0.13 to 0.18 MeV: new ^{13}C evaluation
0.18 to 0.25 MeV: linear interpolation
0.25 to 1.50 MeV: previously evaluated natural carbon data
1.50 to 1.60 MeV: linear interpolation
1.60 to 1.90 MeV: new ^{13}C evaluation
1.90 to 2.00 MeV: linear interpolation
at 2.00 MeV: previously evaluated natural carbon data

The cross sections represented by this file were weighted by the ^{13}C abundance (0.0111) and combined with the previous evaluation with a weight of 0.9889 to generate the final data.

Figures 1, 2 and 3 show the results near the 0.1529 MeV resonance. Figure 1 is for the recommended total cross section of carbon around this resonance for ENDF/B-VI, compared with the data of Heaton et al (5) and the ENDF/B-V evaluation. The peak of this resonance rises above the ENDF/B-V evaluation by 7 %. Figures 2 and 3 display the corresponding Legendre coefficients A_1 and A_2, respectively, also for natural carbon in order to see the relative changes from ENDF/B-V.

Figure 4 shows the total cross section of the 1.736 MeV resonance as would be seen in natural carbon, compared with ENDF/B-V. The peak of the resonance is only 1 % above the background in ENDF/B-V. The s-wave minimum has a 0.2 % effect near 1.72 MeV. Figures 5 and 6 are for the A_1 and A_2 coefficients, respectively, also for natural carbon.

The recommended total elastic cross sections for natural carbon below 2 MeV including the ^{13}C resonances, are listed below. The recommended differential elastic scattering cross sections below 2 MeV are given by

$$\frac{d\sigma}{d\Omega}(E,\Omega) = \frac{\sigma_S(E)}{2\pi} \sum_{l=0}^{L} \frac{2l+1}{2} A_l(E) P_l(m)$$

where the integrated cross sections $\sigma_S(E)$ and the Legendre coefficients $A_l(E)$ are given in the lists of recommended data. Note that $A_0 = 1$. Linear interpolation between the tabulated entries is recommended for intermediate energies.

The energy of the 2078-keV resonance of ^{12}C is a valued energy-scale reference point (see "Neutron Energy Standards" by C Coceva in this volume).

Conclusions and Recommendations

1. The cross sections and angular distributions of ^{13}C near the 0.1529- and 1.736-MeV resonances - ignored in former evaluations because they were based on natural carbon data and because the properties of the 1.736-MeV resonance were then poorly understood - have been evaluated. The results below 2 MeV for natural carbon could be considered for use in high-resolution scattering measurements near these resonances. The largest effect of ^{13}C is in the total cross sections of the 0.1529-MeV resonance. Both the total cross sections of the 1.736-MeV resonance and the angular distributions near each resonance are found to have rather small effects.

2. The 2078.5 ± 0.5 keV resonance energy is a recommended energy calibration (see Neutron Energy Standards by C Coceva in this volume)

References

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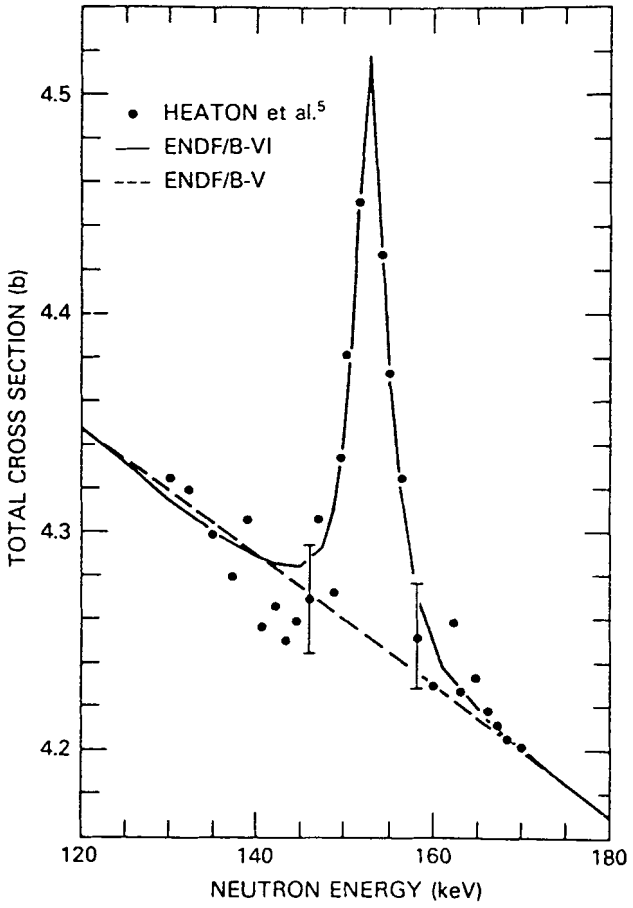

Fig. 1. Total cross sections of natural carbon near the 152.9-keV resonance.

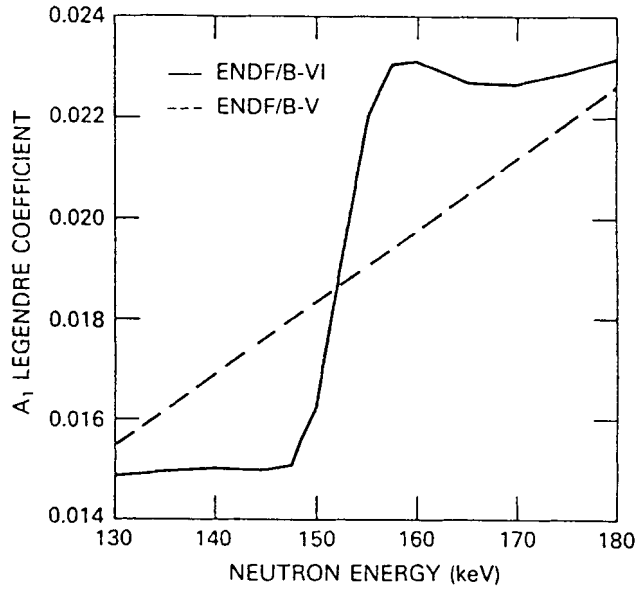

Fig. 2. The A_1 Legendre coefficients of natural carbon near the 152.9-keV resonance.

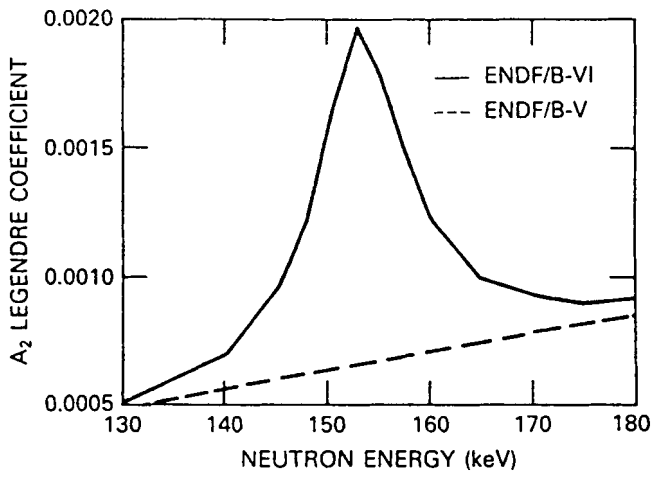

Fig. 3. The A_2 Legendre coefficients of natural carbon near the 152.9-keV resonance.

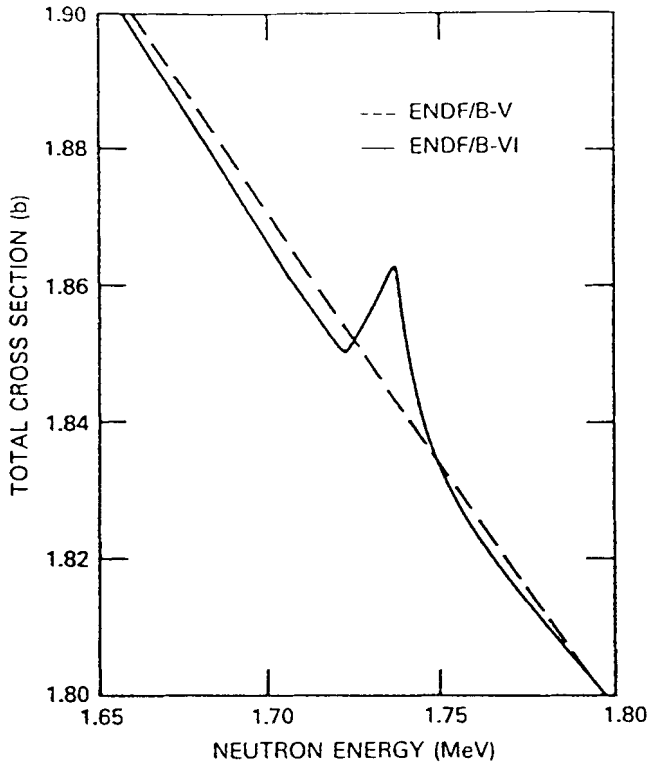

Figure 4 Total cross sections of natural carbon near the 1.736-MeV resonance.

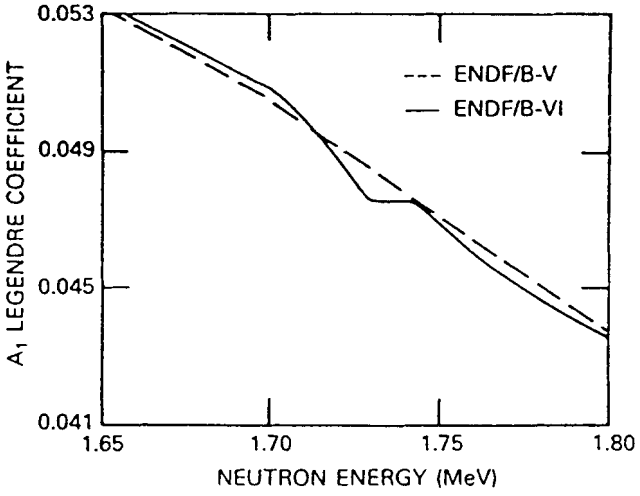

Figure 5 The A_1 Legendre coefficients of natural carbon near the 1.736-MeV resonance. The unusual shape of the results is due to the opposite interference patterns of the two superimposing resonances.

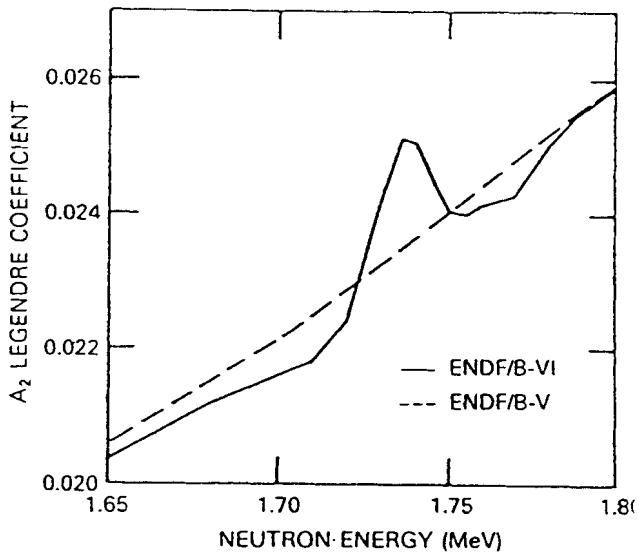

Figure 6 The A_2 Legendre coefficients of natural carbon near the 1.736-MeV resonance.

C(n,n) CROSS SECTIONS - Recommended Reference Data

Numerical values from ENDF/B-VI, MAT-600.
Applicable energy range 1.0E-5 eV to 2 MeV.
Linear-Linear interpolation.

Cross Section Values

E (keV)	XSEC (b)	E (keV)	XSEC (b)	E (keV)	XSEC (b)
1.0000E-08	4.7392	5.0000E00	4.7161	1.0000E01	4.6991
1.5000E00	4.6821	2.0000E01	4.6653	2.5000E01	4.6486
3.0000E01	4.6319	3.5000E01	4.6154	4.0000E01	4.5989
4.5000E01	4.5825	5.0000E01	4.5662	7.5000E01	4.4862
1.0000E02	4.4084	1.2500E02	4.3301	1.3000E02	4.3148
1.3500E02	4.3012	1.4000E02	4.2891	1.4250E02	4.2848
1.4500E02	4.2838	1.4750E02	4.2924	1.4875E02	4.3070
1.5000E02	4.3400	1.5100E02	4.3937	1.5200E02	4.4765
1.5290E02	4.5206	1.5400E02	4.4545	1.5500E02	4.3731
1.5600E02	4.3210	1.5800E02	4.2710	1.6000E02	4.2484
1.6125E02	4.2393	1.6250E02	4.2320	1.6500E02	4.2204
1.7000E02	4.2021	1.7500E02	4.1861	1.8000E02	4.1712
2.0000E02	4.1159	2.2500E02	4.0484	2.5000E02	3.9828
2.7500E02	3.9182	3.0000E02	3.8551	3.2500E02	3.7937
3.5000E02	3.7338	3.7500E02	3.6753	4.0000E02	3.6182
4.2500E02	3.5626	4.5000E02	3.5082	4.7500E02	3.4551
5.0000E02	3.4033	5.2500E02	3.3527	5.5000E02	3.3032
5.7500E02	3.2549	6.0000E02	3.2076	6.2500E02	3.1615
6.5000E02	3.1163	6.7500E02	3.0722	7.0000E02	3.0290
7.2500E02	2.9868	7.5000E02	2.9454	7.7500E02	2.9050
8.0000E02	2.8654	8.2500E02	2.8267	8.5000E02	2.7888
8.7500E02	2.7517	9.0000E02	2.7154	9.2500E02	2.6798
9.5000E02	2.6450	9.7500E02	2.6108	1.0000E03	2.5774
1.0250E03	2.5446	1.0500E03	2.5125	1.0750E03	2.4811
1.1000E03	2.4503	1.1250E03	2.4201	1.1500E03	2.3905
1.1750E03	2.3615	1.2000E03	2.3331	1.2250E03	2.3052
1.2500E03	2.2779	1.2750E03	2.2511	1.3000E03	2.2249
1.3250E03	2.1991	1.3500E03	2.1739	1.3750E03	2.1492
1.4000E03	2.1250	1.4250E03	2.1012	1.4500E03	2.0780
1.4750E03	2.0552	1.5000E03	2.0328	1.5250E03	2.0105
1.5500E03	1.9888	1.5750E03	1.9674	1.6000E03	1.9465
1.6250E03	1.9259	1.6500E03	1.9059	1.6750E03	1.8858
1.6800E03	1.8819	1.7000E03	1.8655	1.7100E03	1.8574
1.7150E03	1.8537	1.7200E03	1.8507	1.7250E03	1.8501
1.7300E03	1.8547	1.7320E03	1.8583	1.7340E03	1.8616
1.7360E03	1.8625	1.7380E03	1.8598	1.7400E03	1.8544
1.7450E03	1.8408	1.7500E03	1.8327	1.7550E03	1.8280
1.7600E03	1.8244	1.7700E03	1.8180	1.7750E03	1.8147

Cross Section Values (cont.)

E (keV)	XSEC (b)	E (keV)	XSEC (b)	E (keV)	XSEC (b)
1.7800E03	1.8115	1.7900E03	1.8048	1.8000E03	1.7980
1.8200E03	1.7844	1.8250E03	1.7810	1.8500E03	1.7644
1.8750E03	1.7480	1.9000E03	1.7319	1.9250E03	1.7161
1.9500E03	1.7006	1.9750E03	1.6852	1.9830E03	1.6807
1.9910E03	1.6755	2.0000E03	1.6704		

Uncertainties in total elastic scattering cross section

Energy range keV	Uncertainties percent
1 - 500	0.46
500 - 1500	0.53
1500-2000	0.60

Centre-of-Mass Legendre Coefficients for Elastically Scattered Neutrons

The recommended differential elastic scattering cross sections below 2 MeV are given by

$$\frac{d\sigma}{d\Omega}(E,\Omega) = \frac{\sigma_S(E)}{2\pi} \sum_{l=0}^{L} \frac{2l+1}{2} A_l(E) P_l(m)$$

where the integrated cross sections $\sigma_S(E)$ are listed in the above table. Note that $A_0 = 1$. Linear interpolation between the tabulated entries is recommended for intermediate energies

E (keV)	A(1)	A(2)	A(3)	A(4)	A(5)
1.000E00	1.4011E-04				
5.000E00	6.9820E-04				
1.000E01	1.3720E-03				
5.000E01	6.6030E-03	7.4231E-05			
1.000E02	1.2457E-02	2.8013E-04			
1.100E02	1.3396E-02	3.5243E-04			
1.200E02	1.4247E-02	4.2976E-04			
1.300E02	1.4911E-02	5.2400E-04			
1.400E02	1.5120E-02	7.0387E-04			
1.450E02	1.4993E-02	9.5793E-04			
1.475E02	1.5160E-02	1.2253E-03			
1.500E02	1.6272E-02	1.6354E-03			
1.529E02	1.9677E-02	1.9837E-03			
1.550E02	2.2008E-02	1.8453E-03			

Centre-of-Mass Legendre Coefficients for Elastically Scattered Neutrons (cont.)

E (keV)	A(1)	A(2)	A(3)	A(4)	A(5)
1.575E02	2.3056E-02	1.5056E-03			
1.600E02	2.3114E-02	1.2506E-03			
1.650E02	2.2837E-02	1.0133E-03			
1.700E02	2.2787E-02	9.3711E-04			
1.750E02	2.2941E-02	9.1774E-04			
1.800E02	2.3220E-02	9.2212E-04			
1.900E02	2.3976E-02	9.6019E-04			
2.000E02	2.4868E-02	1.0143E-03			
3.000E02	3.3897E-02	1.9465E-03	6.3344E-05		
4.000E02	4.1486E-02	3.0116E-03	1.2810E-04		
5.000E02	4.7802E-02	4.1326E-03	2.2308E-04		
6.000E02	5.3145E-02	5.2740E-03	3.4767E-04		
7.000E02	5.7448E-02	6.3949E-03	4.9740E-04	-5.6411E-05	
8.000E02	6.0789E-02	7.4861E-03	6.6509E-04	-1.0135E-04	
9.000E02	6.3226E-02	8.5536E-03	8.4019E-04	-1.7006E-04	
1.000E03	6.4797E-02	9.6181E-03	1.0081E-03	-2.7049E-04	
1.100E03	6.5500E-02	1.0709E-02	1.1468E-03	-4.1188E-04	
1.200E03	6.5360E-02	1.1881E-02	1.2347E-03	-6.0508E-04	
1.300E03	6.4363E-02	1.3199E-02	1.2405E-03	-8.6203E-04	
1.400E03	6.2475E-02	1.4752E-02	1.1228E-03	-1.1952E-03	6.1145E-05
1.500E03	5.9635E-02	1.6643E-02	8.2598E-04	-1.6157E-03	8.8288E-05
1.600E03	5.5781E-02	1.8970E-02	2.4924E-04	-2.1263E-03	1.2479E-04
1.650E03	5.3285E-02	2.0395E-02	-2.5520E-04	-2.4161E-03	1.4880E-04
1.680E03	5.1806E-02	2.1170E-02	-5.9070E-04	-2.5785E-03	1.6321E-04
1.700E03	5.0765E-02	2.1580E-02	-8.6788E-04	-2.6585E-03	1.7282E-04
1.710E03	4.9951E-02	2.1849E-02	-1.1148E-03	-2.6614E-03	1.7895E-04
1.720E03	4.8786E-02	2.2438E-02	-1.3869E-03	-2.5818E-03	1.8508E-04
1.730E03	4.7608E-02	2.4263E-02	-1.4102E-03	-2.4353E-03	1.9121E-04
1.736E03	4.7630E-02	2.5140E-02	-1.1949E-03	-2.4536E-03	1.9489E-04
1.740E03	4.7709E-02	2.5087E-02	-1.0515E-03	-2.5254E-03	1.9734E-04
1.745E03	4.7471E-02	2.4548E-02	-9.6987E-04	-2.6608E-03	2.0040E-04
1.750E03	4.6930E-02	2.4108E-02	-1.0242E-03	-2.8044E-03	2.0347E-04
1.755E03	4.6394E-02	2.4023E-02	-1.1503E-03	-2.9172E-03	2.0653E-04
1.760E03	4.5964E-02	2.4154E-02	-1.2868E-03	-2.9973E-03	2.0960E-04
1.770E03	4.5291E-02	2.4589E-02	-1.5293E-03	-3.1044E-03	2.1572E-04
1.780E03	4.4688E-02	2.5035E-02	-1.7368E-03	-3.1831E-03	2.2185E-04
1.790E03	4.4087E-02	2.5456E-02	-1.9256E-03	-3.2520E-03	2.2798E-04
1.800E03	4.3475E-02	2.5860E-02	-2.1044E-03	-3.3168E-03	2.3411E-04
1.820E03	4.1905E-02	2.6777E-02	-2.6118E-03	-3.4112E-03	2.4827E-04
1.850E03	3.9503E-02	2.8128E-02	-3.3554E-03	-3.5484E-03	2.6950E-04
1.900E03	3.4624E-02	3.0586E-02	-5.0853E-03	-3.5966E-03	3.0656E-04
1.925E03	3.1690E-02	3.1891E-02	-6.2675E-03	-3.4744E-03	3.2456E-04
1.950E03	2.8250E-02	3.3253E-02	-7.8156E-03	-3.1577E-03	3.4068E-04
1.960E03	2.6670E-02	3.3818E-02	-8.5857E-03	-2.9434E-03	3.4610E-04
1.970E03	2.4929E-02	3.4405E-02	-9.4769E-03	-2.6547E-03	3.5055E-04
1.980E03	2.2980E-02	3.5026E-02	-1.0524E-02	-2.2657E-03	3.5365E-04

Centre-of-Mass Legendre Coefficients for Elastically Scattered Neutrons (cont.)

E (keV)	A(1)	A(2)	A(3)	A(4)	A(5)
1.990E03	2.0759E-02	3.5700E-02	-1.1780E-02	-1.7369E-03	3.5478E-04
2.000E03	1.8161E-02	3.6480E-02	-1.3322E-02	-1.0069E-03	3.5306E-04

THE ^{197}Au (n,γ) ^{198}Au CROSS SECTION

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The radiative neutron capture cross section of gold can be determined by two different methods: activation and prompt γ-ray detection. The first method is of straightforward application since the product nucleus formed by neutron capture in this monoisotopic element has a simple and well known decay scheme. In this case, the main factor limiting the accuracy of the measurements is the influence of back-ground neutrons from various sources. Prompt γ-ray detection associated with a white neutron source and a precise determination of the relative neutron flux has the advantage of allowing a self-calibration at thermal or at low energy resonances. Above 250 keV, however, γ-rays produced by inelastic neutron scattering mask the low energy portion of the gamma spectrum. Extrapolation of the spectrum to zero pulse height introduces a systematic uncertainty which increases with energy. Also, in the case of prompt γ-ray detection, the following warning should be issued: full use of the standard and of its inherent precision can only be made if the detector employed has an efficiency completely independent of the capture γ-ray spectrum shape. This is not always the case as shown by the difficulties encountered in recent years in the study of ^{56}Fe capture with total energy detectors (1). Although the situation is now considerably improved (2), caution is still necessary.

The cross section is considered standard in the range 0.2-3.5 MeV: there is no reason why this region should not be extended to lower energy provided the single data points are replaced by averages in appropriate energy intervals.

Works published from 1982 on, and therefore not covered in the previous edition of this report (3), are briefly described in the following. One may notice that the activity in the field is still considerable although some papers do not present completely original data but are rather the finalization of experiments already reported previously.

Status and Recent Results

1) Yamamuro et al. (4) determined the cross section in the range 3.2-270 keV using a white neutron source and measuring the prompt γ-rays with two C_6D_6 detectors. Neutron flux was measured with the reaction $^{10}B(n,\alpha\gamma)$ taking ENDF/B-V as a standard. Relative data were normalized to the average capture cross section at 24 keV measured in an absolute way using an iron-filtered neutron beam and found equal to 640 ± 35 mb. Its 5.5% relative error represents the main contribution to the uncertainties of the data. It is a pity

that the calibration was not checked, and perhaps improved in accuracy, at the saturated 4.9 eV resonance. The results are reported in Table I.

2) Hussain and Hunt (5) performed absolute activation measurements in the range 2-3.5 MeV. The neutron flux produced by the reaction D(d, n)^{3}He was estimated by detecting protons from the competing D(d, p)^{3}H reaction. Using a 2 MeV deuteron beam, the neutron energy was varied by changing the angle subtended by the sample at the target with respect to the direction of the incident beam. Corrections for the angular distribution of the (d, n) and (d, p) reactions were introduced. The estimated overall uncertainty of the data, listed in Table II, is 8.5%, by far the largest contribution to it is given by the correction for the activity induced by scattered neutrons. The results are given in Table II.

3) Andersson et al. (6) measured the activation cross section in the range 2.0-7.6 MeV relative to that of the reaction ^{115}In(n, n') ^{115m}In. This reference cross section was chosen because it is almost constant with neutron energy over the entire range and it is known with good accuracy (6-10%). The influence of the background neutrons produced either by charged particle reactions in the target or by nonelastic reactions in the sample and surrounding materials were investigated in great detail. The data are listed in Table III.

4) Kazakov et al. (7) determined the cross section in the range 3-420 keV by measuring prompt γ-rays with a liquid scintillator tank at the Obninsk pulsed Van de Graaff accelerator. Neutron flux was determined using the reactions ^{6}Li(n,α) below 100 keV and ^{10}B(n,$\alpha\gamma$) above 100 keV. Data were normalized with the saturated resonance method. In Table IV are listed the values and associated uncertainties for the region above 100 keV.

5) Joly (8) used the same spectrum fitting method developed for the determination of σ(n,γ) to derive the cross section associated with the (n,γn') process. He showed that the importance of this reaction increases with energy until it becomes comparable with σ(n,γ). For example at 3 MeV neutron energy about 32% of the primary γ-spectrum populates unbound levels which decay subsequently mainly by neutron emission. The final data from Voignier et al are listed in Table V for the energy range 0.5-3.0 MeV.

6) Davletshin et al. (9) measured the activation cross section pointwise relative to the H(n, n) cross section in the range 164-1389 keV, The data, listed in Table VI, are partly unpublished and partly previously published results to which a correction for the anisotropy and non-monochromaticity of the neutron source was applied. This correction produced a systematic decrease of the measured cross sections by an amount of 5 to 9 % for E < 550 keV and 2 % for 1000 < E < 1200 keV. The averaged relative deviations of these data from the ENDF/B-VI evaluation are included in a band going from + 7 % to - 4 % except for one point at 166 keV which is 18 % higher.

7) Demekhin et al (10) measured the activation cross section at E = 2.7 MeV and found the value $\sigma g = 33 \pm 4$ mb

8) Ratynski and Käppeler (11) measured the activation cross section for the quasistellar neutron spectrum at kT = 25 keV provided by the $^7Li(p,n)^7Be$ reaction. They showed that this spectrum follows the Maxwellian shape down to 1 keV neutron energy and that it is not very sensitive to the proton energy. A Maxwellian averaged cross section value $\langle sv \rangle / n_T = 648 \pm 10$ mb was derived, and extrapolation to the common s-process temperature kT = 30 keV yielded 582 ± 9 mb. A comparison with the Maxwellian averaged cross section calculated from the differential data of Macklin (12) allowed to obtain a renormalization factor for these data:

$$R_{norm} = [(648 \pm 10)\ \text{mb}]/655\ \text{mb} = 0.989 \pm 1.5\ \%$$

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Table I. Data from Yamamuro et al. (4).

Energy Range (keV)	Cross section (mb)
3.2 - 4	2830 ± 176
4 - 5	2260 ± 129
5 - 6	1890 ± 109
6 - 8	1620 ± 92
8 - 10	1250 ± 72
10 - 12	1130 ± 65
12 - 15	923 ± 53
15 - 18	768 ± 45
18 - 21	681 ± 40
21 - 25	638 ± 37
25 - 30	592 ± 34
30 - 35	537 ± 32
35 - 45	487 ± 28
45 - 55	419 ± 24
55 - 65	369 ± 21
65 - 80	360 ± 21
80 - 100	318 ± 18
100 - 120	303 ± 17
120 - 150	275 ± 16
150 - 180	265 ± 15
180 - 220	256 ± 15
220 - 270	246 ± 14

Table II. Data from Hussain et al. (5).

Neutron Energy (MeV)	Cross Section[a)] (mb)
2.12	53.9
2.27	45.2
2.35	41.0
2.44	37.1
2.87	26.2

a) relative error equal to 8.5 % throughout

Table III. Data from Andersson et al. (6)

Neutron Energy (Mev)	Cross section (mb)
2.03	53 ± 6
2.26	46 ± 6
2.54	37 ± 5
2.68	25 ± 3
2.93	21.7 ± 2.4
3.17	17.8 ± 1.6
3.42	16.3 ± 1.5
3.69	14.8 ± 1.3
3.93	13.2 ± 1.2
4.13	11.8 ± 1.6
4.18	11.6 ± 1.1
4.59	8.8 ± 1.1
4.87	8.5 ± 1.5
5.26	6.9 ± 0.8
5.58	6.2 ± 0.9
5.90	5.1 ± 0.8
6.25	4.7 ± 0.7
6.49	4.3 ± 0.9
6.85	2.4 ± 0.5
7.13	2.3 ± 0.6
7.26	1.3 ± 0.5
7.60	0.8 ± 0.5

Table IV. Data from Kazakov et al. (7).

Energy range (keV)	Cross section (mb)
100 - 110	322 ± 14
110 - 120	296 ± 13
120 - 130	285 ± 12
130 - 140	283 ± 12
140 - 150	277 ± 12
150 - 160	281 ± 12
160 - 170	276 ± 12
170 - 180	271 ± 12
180 - 190	266 ± 11
190 - 200	260 ± 11
200 - 210	261 ± 11
210 - 220	269 ± 11
220 - 230	269 ± 11
230 - 240	258 ± 11
240 - 250	257 ± 11
250 - 260	250 ± 11
260 - 270	249 ± 11
270 - 280	247 ± 11
280 - 290	217 ± 9
290 - 300	209 ± 9
300 - 320	204 ± 9
320 - 340	189 ± 8
340 - 360	175 ± 8
360 - 380	163 ± 7
380 - 400	155 ± 7
400 - 420	151 ± 6

Table V Data from Voignier et al (8)

Neutron Energy (MeV)	Cross section (mb)
0.50	130 ± 9
0.70	101 ± 7
1.00	79 ± 8
2.00	52 ± 8
2,50	39.7 ± 5
3.00	26.8 ± 3.5

Table VI Data from Davletshin et al (9)

Neutron Energy (keV)	Cross section (mb)
164 ± 23	275 ± 5 %
166 ± 22	312 ± 5 %
166 ± 22	275 ± 5 %
336 ± 17	193 ± 4 %
340 ± 17	198 ± 4 %
340 ± 17	185 ± 4 %
342 ± 10	198 ± 4 %
344 ± 24	180 ± 4 %
344 ± 24	185 ± 4 %
467 ± 21	142 ± 4 %
471 ± 22	144 ± 4 %
475 ± 21	140 ± 4 %
544 ± 27	120 ± 4 %
544 ± 28	118 ± 4 %
619 ± 35	107 ± 4 %
627 ± 34	100 ± 4 %
630 ± 37	106 ± 4 %
636 ± 34	100 ± 4 %
636 ± 21	101 ± 4 %
637 ± 20	100 ± 4 %
637 ± 20	106 ± 4 %
645 ± 23	102 ± 4 %
693 ± 22	118 ± 4 %
708 ± 22	97 ± 4 %
710 ± 23	90 ± 4 %
764 ± 34	92 ± 4%
779 ± 31	86 ± 4 %
779 ± 31	86 ± 4 %
780 ± 31	91 ± 4 %
881 ± 21	94 ± 4 %
883 ± 17	91 ± 4 %
885 ± 21	90 ± 4 %
1037 ± 18	81 ± 4 %
1042 ± 22	80 ± 4 %
1133 ± 20	83 ± 4 %
1135 ± 21	81 ± 4 %
1142 ± 17	80 ± 4 %
1181 ± 20	74 ± 4 %
1192 ± 15	72 ± 4 %
1389 ± 17	68 ± 4 %

^{197}Au(n,γ)^{198}Au CROSS SECTIONS - Recommended Reference Data

Numerical values from ENDF/B-VI, MAT-7925
Applicable energy range 0.2 to 2.5 MeV
Linear-linear interpolation

Cross section values

E(keV)	XSEC(b)	E(keV)	XSEC(b)	E(keV)	XSEC(b)
1.90E02	2.534E-01	2.00E02	2.502E-01	2.10E02	2.469E-01
2.20E02	2.445E-01	2.30E02	2.415E-01	2.35E02	2.403E-01
2.40E02	2.388E-01	2.45E02	2.374E-01	2.50E02	2.360E-01
2.60E02	2.331E-01	2.70E02	2.299E-01	2.80E02	2.141E-01
3.00E02	1.999E-01	3.25E02	1.877E-01	3.50E02	1.778E-01
3.75E02	1.689E-01	4.00E02	1.614E-01	4.25E02	1.538E-01
4.50E02	1.462E-01	4.75E02	1.389E-01	5.00E02	1.324E-01
5.20E02	1.270E-01	5.40E02	1.236E-01	5.70E02	1.186E-01
6.00E02	1.084E-01	6.50E02	1.002E-01	7.00E02	9.640E-02
7.50E02	9.280E-02	8.00E02	8.970E-02	8.50E02	8.690E-02
9.00E02	8.430E-02	9.40E02	8.250E-02	9.60E02	8.180E-02
9.80E02	8.100E-02	1.00E03	8.030E-02	1.10E03	7.720E-02
1.25E03	7.290E-02	1.40E03	6.940E-02	1.60E03	6.650E-02
1.80E03	5.960E-02	2.00E03	5.340E-02	2.20E03	4.330E-02
2.40E03	3.600E-02	2.50E03	3.340E-02		

Uncertainties

Reviewers of the ENDF/B-VI standards cross sections which were evaluated by the US Cross Section and Evaluation Group (CSEWG) (see Introduction) have expressed the concern that the uncertainties resulting from the combination of R-matrix and simultaneous evaluations might be too small. As a result, the CSEWG Standards Subcommittee at its May 1990 Meeting produced a set of expanded covariance estimates for the standard cross section reactions. These uncertainties are estimates such that if a modern day experiment were performed on a given standard cross section using the best techniques, approximately 2/3 of the results should fall within these expanded uncertainties. The expanded uncertainties for the ^{197}Au(n,γ)^{198}Au cross section are given in the following table and are compared to values from the combined output of the standards covariance analysis.

Energy Range (keV)	Expanded Uncertainty (percent)	Combined Analysis (percent)
2.53E-05	0.14	0.14
200-500	3.0	1.31
500-1000	3.5	2.1
1000-2500	4.5	2.0

THE ^{235}U FISSION CROSS SECTION

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Description

The ^{235}U fission cross-section is a good standard over the energy range 100 keV to 20 MeV and even higher (1) because

(a) the fission process has a high Q-value;
(b) the cross-section is of reasonable magnitude at all energies of interest;
(c) ^{235}U is very suitable for use in fission chambers as it is readily obtainable, fission foils can be assayed to high accuracy (2) and it has a long half-life so minimising pulse pile-up and handling problems.

Though the cross-section is recommended for use as a standard above 100 keV care has to be taken in its use because of the known structure in the cross-section between 100 and ~300 keV.

Status

The ^{235}U fission cross-section is one of the standards which have been evaluated simultaneously for ENDF/B-VI (3,4) by a committee of the U.S. Cross-section Evaluation Working Group consisting of A. Carlson (Chairman), G. Hale, W.P. Poenitz and R. Peelle. In this the cross-sections were determined for $^{6}Li(n,\alpha)$, $^{10}B(n,\alpha_0)$, $^{10}B(n,\alpha_1)$, $^{197}Au(n,\gamma)$ and $^{235}U(n,f)$ along with the important cross-sections $^{238}U(n,\gamma)$, $^{238}U(n,f)$ and $^{239}Pu(n,f)$. The goals were to correctly utilise all types of absolute and ratio data while taking full advantage of the R-matrix representation for the light-element standards. R-matrix analyses of Hale were used for the ^{7}Li and ^{11}B systems, and Poenitz applied his simultaneous least-squares fitting system to the data sets that are independent of those used in the R-matrix fits. The 2.2 km/s results of E.J. Axton (5) were included with the experimental pointwise data. The outputs of these studies, including their full covariance matrices, were combined by Peelle in a single least-squares adjustment. The original full data base was not internally consistent within the quoted uncertainties.

The resulting fission cross-section data for ^{235}U are plotted in Fig. 1 and are given below together with uncertainties. Also shown are two recent measurements (6,7). Other measurements where the data are only available in graphical form (8,9) support the evaluation up to 15 MeV but indicate that at higher energies the evaluation is too low by up to 5 %. Fig 2 shows the ratio of the cross-section to that recommended in ENDF/B-V.

Comments and Recommendation

It is believed that the proposed ENDF/B-VI evaluation is the best that can be performed given the discrepant data base unless much additional work is performed to try to identify the additional errors in the available measurement. At 14.70 MeV it is supported by the independent simultaneous evaluation of Ryves (10) which is based on 9 activation cross-sections in addition to the H(n,n), ^{235}U(n,f) and ^{238}U(n,f) reactions. It gives a value of 2.091 ± 0.7 % which is in excellent agreement with the ENDF/B-VI value of 2.0836. It is therefore recommended that the ENDF/B-VI evaluation for the ^{235}U fission cross section should be used as the standard reference data.

If the errors of the ENDF/B-VI evaluation are as given in the combined analysis the evaluated cross-section meets the requirements for 1% accuracy specified in WRENDA 83/84 over most of the energy range. However, it is felt that the expanded uncertainties are more realistic. That means that additional measurements are still required but it is recommended that these should only be performed if they meet one or more of the following objectives:

(a) they have an accuracy approaching that associated with the evaluation;
(b) they identify errors in existing techniques or measurements;
(c) they give data in energy ranges where the present data base is weak (i.e. 6-14 MeV and 15-20 MeV);
(d) they use a new technique.

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Figure 1

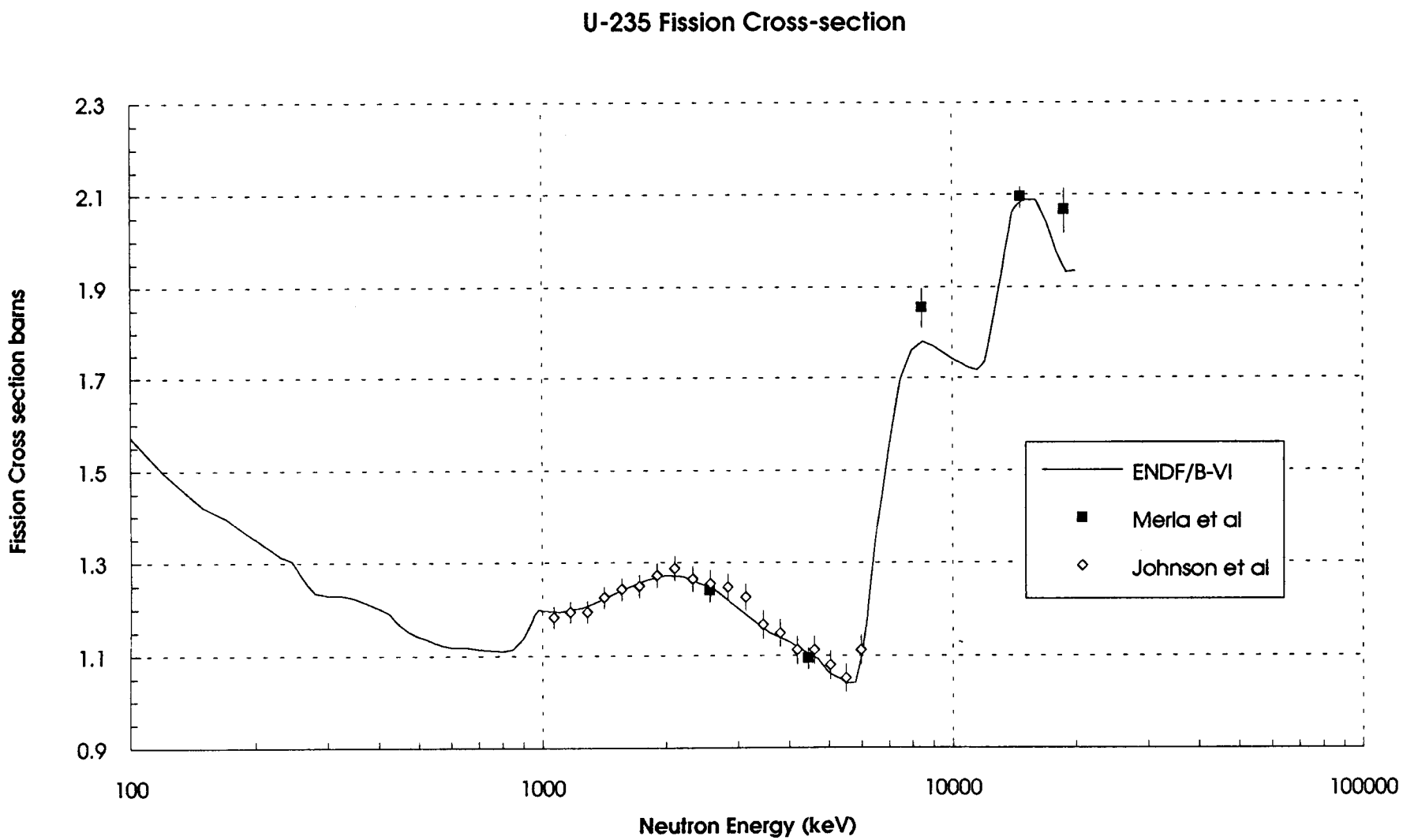
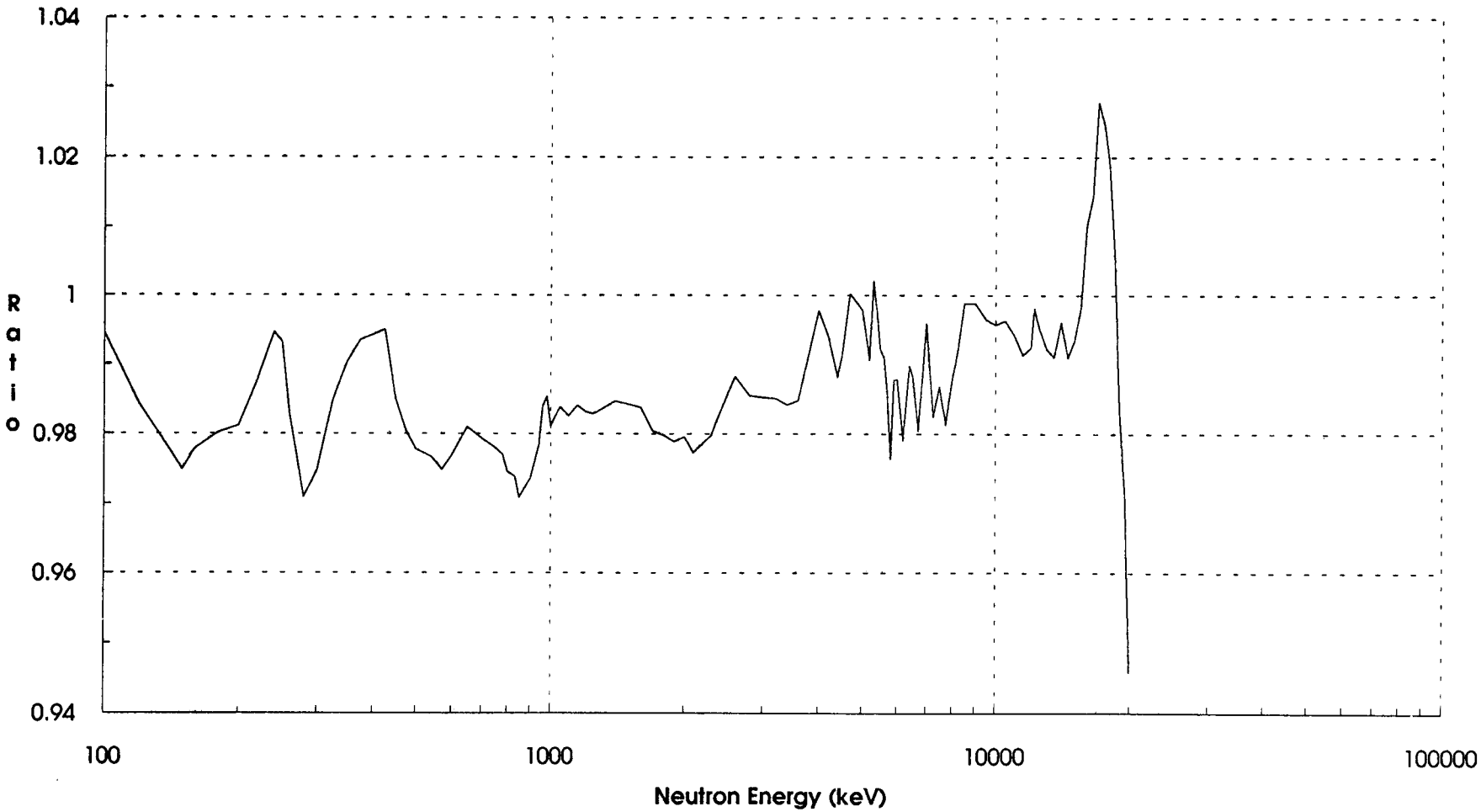

U-235 Fission Cross-section
Fission Cross section barns
2.3
2.1
1.9
1.7
1.5
1.3
1.1
0.9
100
1000
10000
100000
Neutron Energy (keV)
ENDF/B-VI
Merla et al
Johnson et al

Figure 2

Ratio of ENDF/B-VI Fission Cross-section to ENDF/B-V



^{235}U FISSION CROSS SECTIONS - Recommended Reference Data

Numerical values from ENDF/B-VI, MAT 9228
Applicable energy range 0.1-20.0 MeV
Linear-linear interpolation

Cross section values

E(keV)	XSEC(b)	E(keV)	XSEC(b)	E(keV)	XSEC(b)
1.000E+02	1.5724E+00	1.034E+02	1.5580E+00	1.100E+02	1.5320E+00
1.200E+02	1.4961E+00	1.299E+02	1.4689E+00	1.400E+02	1.4433E+00
1.500E+02	1.4203E+00	1.512E+02	1.4189E+00	1.600E+02	1.4081E+00
1.700E+02	1.3967E+00	1.714E+02	1.3942E+00	1.721E+02	1.3930E+00
1.800E+02	1.3800E+00	1.900E+02	1.3647E+00	1.980E+02	1.3537E+00
2.000E+02	1.3510E+00	2.100E+02	1.3370E+00	2.200E+02	1.3265E+00
2.250E+02	1.3197E+00	2.264E+02	1.3178E+00	2.300E+02	1.3130E+00
2.350E+02	1.3100E+00	2.400E+02	1.3070E+00	2.450E+02	1.3030E+00
2.500E+02	1.2930E+00	2.502E+02	1.2926E+00	2.600E+02	1.2690E+00
2.700E+02	1.2500E+00	2.750E+02	1.2424E+00	2.800E+02	1.2350E+00
2.927E+02	1.2318E+00	2.960E+02	1.2310E+00	3.000E+02	1.2300E+00
3.250E+02	1.2300E+00	3.342E+02	1.2274E+00	3.400E+02	1.2258E+00
3.500E+02	1.2230E+00	3.588E+02	1.2194E+00	3.687E+02	1.2155E+00
3.750E+02	1.2130E+00	3.949E+02	1.2042E+00	4.000E+02	1.2020E+00
4.166E+02	1.1940E+00	4.250E+02	1.1900E+00	4.285E+02	1.1865E+00
4.318E+02	1.1833E+00	4.404E+02	1.1751E+00	4.476E+02	1.1684E+00
4.500E+02	1.1662E+00	4.750E+02	1.1510E+00	4.763E+02	1.1505E+00
5.000E+02	1.1410E+00	5.200E+02	1.1365E+00	5.400E+02	1.1300E+00
5.500E+02	1.1273E+00	5.524E+02	1.1266E+00	5.700E+02	1.1220E+00
6.000E+02	1.1185E+00	6.500E+02	1.1182E+00	6.528E+02	1.1179E+00
7.000E+02	1.1135E+00	7.500E+02	1.1120E+00	7.733E+02	1.1111E+00
8.000E+02	1.1100E+00	8.500E+02	1.1135E+00	9.000E+02	1.1372E+00
9.039E+02	1.1403E+00	9.400E+02	1.1691E+00	9.600E+02	1.1876E+00
9.800E+02	1.1992E+00	1.000E+03	1.1969E+00	1.044E+03	1.1955E+00
1.090E+03	1.1941E+00	1.100E+03	1.1938E+00	1.200E+03	1.1994E+00
1.250E+03	1.2020E+00	1.300E+03	1.2082E+00	1.336E+03	1.2125E+00
1.400E+03	1.2200E+00	1.436E+03	1.2245E+00	1.500E+03	1.2321E+00
1.600E+03	1.2435E+00	1.700E+03	1.2529E+00	1.750E+03	1.2575E+00
1.800E+03	1.2619E+00	2.000E+03	1.2714E+00	2.200E+03	1.2699E+00
2.400E+03	1.2561E+00	2.500E+03	1.2500E+00	2.600E+03	1.2442E+00
2.800E+03	1.2220E+00	3.000E+03	1.2010E+00	3.500E+03	1.1554E+00
3.600E+03	1.1473E+00	4.000E+03	1.1295E+00	4.500E+03	1.1011E+00
4.700E+03	1.0923E+00	5.000E+03	1.0617E+00	5.250E+03	1.0521E+00
5.300E+03	1.0502E+00	5.321E+03	1.0490E+00	5.500E+03	1.0388E+00
5.750E+03	1.0405E+00	5.800E+03	1.0408E+00	6.000E+03	1.0985E+00
6.200E+03	1.1817E+00	6.250E+03	1.2085E+00	6.500E+03	1.3481E+00
6.750E+03	1.4458E+00	7.000E+03	1.5467E+00	7.250E+03	1.6211E+00
7.500E+03	1.6964E+00	7.750E+03	1.7300E+00	8.000E+03	1.7606E+00

E(keV)	XSEC(b)	E(keV)	XSEC(b)	E(keV)	XSEC(b)
8.250E+03	1.7704E+00	8.500E+03	1.7800E+00	8.750E+03	1.7749E+00
9.000E+03	1.7700E+00	9.250E+03	1.7628E+00	9.500E+03	1.7558E+00
9.750E+03	1.7485E+00	1.000E+04	1.7415E+00	1.025E+04	1.7365E+00
1.050E+04	1.7317E+00	1.075E+04	1.7267E+00	1.100E+04	1.7219E+00
1.125E+04	1.7194E+00	1.150E+04	1.7170E+00	1.175E+04	1.7259E+00
1.200E+04	1.7347E+00	1.219E+04	1.7668E+00	1.225E+04	1.7760E+00
1.250E+04	1.8175E+00	1.275E+04	1.8588E+00	1.300E+04	1.9002E+00
1.325E+04	1.9401E+00	1.350E+04	1.9801E+00	1.375E+04	2.0200E+00
1.400E+04	2.0600E+00	1.425E+04	2.0701E+00	1.450E+04	2.0800E+00
1.475E+04	2.0845E+00	1.500E+04	2.0890E+00	1.525E+04	2.0890E+00
1.550E+04	2.0890E+00	1.575E+04	2.0890E+00	1.600E+04	2.0890E+00
1.650E+04	2.0652E+00	1.700E+04	2.0413E+00	1.750E+04	2.0081E+00
1.796E+04	1.9773E+00	1.800E+04	1.9748E+00	1.850E+04	1.9537E+00
1.900E+04	1.9325E+00	1.950E+04	1.9334E+00	2.000E+04	1.9343E+00

Thermal cross section (2200 m/s) at 0K = 584.25 b

Uncertainties

Reviewers of the ENDF/B-VI standards cross sections which were evaluated by the US Cross Section and Evaluation Group (CSEWG) (see Introduction) have expressed the concern that the uncertainties resulting from the combination of R-matrix and simultaneous evaluations might be too small. As a result, the CSEWG Standards Subcommittee at its May 1990 Meeting produced a set of expanded covariance estimates for the standard cross section reactions. These uncertainties are estimates such that if a modern day experiment were performed on a given standard cross section using the best techniques, approximately 2/3 of the results should fall within these expanded uncertainties. The expanded uncertainties for the ^{235}U fission cross section are given in the following table and are compared to values from the combined output of the standards covariance analysis.

Energy Range (keV)	Expanded Uncertainty (percent)	Combined Analysis (percent)
2.53E-05	0.2	0.19
150-600	1.5	
600-1000	1.6	0.60
1000-3000	1.8	
3000-6000	2.3	0.69
6000-10000	2.2	
10000-12000	1.8	1.14
12000-14000	1.2	
14000-14500	0.8	0.55
14500-15000	1.5	
15000-16000	2.0	0.97

Energy Range (keV)	Expanded Uncertainty (percent)	Combined Analysis (percent)
16000-17000	2.5	
17000-19000	3.0	1.26
19000-20000	4.0	

THE ^{235}U FISSION FRAGMENT ANISOTROPY

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Introduction

For the evaluation of fission fragment cross section measurements, which cover only a small range of angles, the knowledge of fission fragment anisotropies is important. Fission fragment anisotropies contain also essential information about the quantum states available at the saddle point of the fissioning nucleus and these in turn provide a better theoretical understanding of the fission cross section.

Since the last reviews in 1983 (ref. (1)) and 1984 (ref. (2)) of the status of the measured fission fragment anisotropies as function of incident neutron energy in ^{235}U only a few new measurements have been performed. It has been mentioned by Sowerby et al. (ref. (2)) that differences in the fission fragment anisotropies make only small corrections to the measured fission cross section, because virtually all experimental investigations collect fission fragments in 2π geometry. However, even for these investigations the fragment angular distribution is needed to correct for the self absorption within the target for small emission angles with respect to the plane of the deposit. Also from a physics point of view it is desirable to improve the knowledge of fission fragment anisotropies over the whole range of energies under investigation.

Description of the data

In the last decade Frisch gridded ionization chambers have become widely used for measurements of fission fragment properties, in particular angular distributions. This technique has been well established for high resolution neutron induced and spontaneous fission fragment studies (see e.g. ref. (3)). It has an advantageous $2*2\pi$ geometry and the fission fragment angle q with respect to the symmetry axis of the chamber can be measured with a resolution in $\cos q < 0.05$.

Two experiments focused on the neutron energy range from 3 keV to 500 keV (ref. (4)) and from 0.5 MeV to 6 MeV in steps of 0.5 MeV (ref. (5)) have been performed with the same Frisch gridded ionization chamber technique (see ref. (3)). The results are discussed in the next chapter.

Furthermore, investigations have been focused on the energy range of the (n, 2nf) threshold at about 14 MeV incident neutron energy (refs. (6, 7)). These values have been put together in Table 1. It is evident that both authors

disagree with previously measured anisotropy values dated before 1982 and compiled in ref. (1), which typically had values of 0.3 (see ref. (1), Fig. 2).

Another investigation covering the incident neutron energy range from 0.014 to 7.15 MeV has been performed by a Russian group of Obninsk (8). The angular anisotropy data of ref. (8) however have not been included in the present review due to the fact that it was not possible to read them from the picture. An attempt was made to get the numerical values directly from D.L. Shpak, but unfortunately the data could not be obtained because the results are still preliminary.

Discussion of the data

The measured fragment anisotropies $A = ((W(0^o) / W(90^o) -1)$ in the neutron energy range 0-7.5 MeV are shown in Fig. 1-4 and given in Table 2. Fig. 1 shows the recently measured anisotropies between 0 and 7.5 MeV from refs. (4, 5) together with those from refs. (9, 10) and the theoretical curve of ref. (1) based on the statistical theory.

Due to the fact that the theoretical prediction and the newest experimental results do agree very well, it is recommended to use the solid line in fig 1 for cross section evaluations at least in the neutron energy range 1 to 6 MeV.

Figs. 2 to 4 are magnifications of the low energy part below 0.5 MeV incident neutron energy and focus on the negative anisotropies measured by various authors (refs. 5, 9, 10, 11). The full line is an eye guide only. The dashed lines are theoretical calculations of the angular distribution based on the statistical theory (13). These calculations assume that the distribution of K, the projection of the total angular momentum J of the compound nucleus on the symmetry axis, is a gaussian centered around K = 0. The K_0^2-value which is indicated on the Figures 2 to 4 are the variance of the K-distributions. The negative anisotropies cannot be reproduced if only axial symmetry at the saddle point is assumed with no restriction on the possible quantum numbers J, K, and the parity π (see Fig. 2). Only if restrictions in the possible sets of quantum numbers are taken into account as in Figs. 3 and 4, the measured negative anisotropies can be reproduced, meaning different shapes of the nucleus at the saddle point. The R operation is a rotation of 180 degrees around an axis perpendicular to the symmetry axis. The S-operator is equal to R*P, with P being the parity operator. The S-operation is therefore a reflection in a plane containing the symmetry axis (see ref. (14)).

However the only difference in the case of low energy neutron induced fission on ^{235}U (E_n < 0.2 MeV) between R- and S-invariance is that for R-invariance and K = 0 a (J, π) = (4,-) state is possible, whereas for S-invariance and K = 0 a (J, π) = (3, -) state is possible. S-invariance would be in agreement with the suggestion of ref. (15) that the nucleus at the saddle point would be pear-shaped. In ref. (16) it is also mentioned that S-invariance and lack of R-invariance might be expected for the nuclear shape at outer barrier deformations.

The long-standing question whether the angular distribution is dependent on the mass of the fission fragment, was taken into consideration in several investigations (4, 17, 18). However due to low statistics the results obtained by different authors are not unequivocal to draw a conclusion on the mass dependence of the angular distribution. In photon induced fission of ^{235}U however, a clear indication of the mass dependence of the angular distribution has been found (19). A mass dependence of the fission fragment anisotropy is expected if different barriers with different quantum states exist. In the recently introduced multi-modal random-neck rupture model (20), different fission paths are proposed with different barrier heights, which lead to different fission configurations. The model has been successfully compared to experimental data of $^{235}U(n,f)$ (21) and other actinides (see (20)). The potential energy landscape calculated within the framework of this model exhibits at least two different barriers, for the so-called standard channel yielding asymmetric masses, and the super-long channel yielding symmetric masses.

The theoretical finding of at least two distinct fission barriers implies a visible effect in the angular distribution as function of fragment mass at least if asymmetric and symmetric fragmentations are compared. This was the case in the photon induced fission investigation (19). The low yield of symmetric fission for incident neutron energies below the second chance fission threshold has been the bottleneck to clarify this question till now for neutron induced fission of ^{235}U.

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E_n[MeV]	A	ref.
14.12	0.441 ± 0.056	(7)
14.63	0.426 ± 0.024	(6)
14.80	0.333 ± 0.022	(7)

Table 1 : Measured fragment anisotropies A = W(0°)/W (90°) -1 around 14 MeV neutron energy

ref. (4)	
E_n[MeV]	Anisotropy A
0.5	0.096 ± 0.019
1.0	0.140 ± 0.014
1.5	0.181 ± 0.012
2.0	0.155 ± 0.014
2.5	0.180 ± 0.015
3.0	0.181 ± 0.018
3.5	0.153 ± 0.016
4.0	0.155 ± 0.015
4.5	0.170 ± 0.019
5.0	0.159 ± 0.018
5.5	0.143 ± 0.022
6.0	0.176 ± 0.018

ref. (5)	
E_n[keV]	A
3 - 10	0.009 ± 0.007
10 - 20	- 0.029 ± 0.005
20 - 30	- 0.027 ± 0.006
30 - 40	- 0.066 ± 0.012
40 - 50	- 0.035 ± 0.014
50 - 60	- 0.072 ± 0.015
60 - 70	- 0.062 ± 0.013
70 - 80	- 0.055 ± 0.017
80 - 90	- 0.079 ± 0.021
90 - 100	- 0.065 ± 0.022
100 - 125	- 0.063 ± 0.012
125 - 150	- 0.022 ± 0.014
150 - 175	- 0.042 ± 0.012
175 - 200	- 0.029 ± 0.017
200 - 250	- 0.038 ± 0.017
250 - 300	- 0.031 ± 0.019
300 - 350	0.017 ± 0.015
350 - 400	0.035 ± 0.015
400 - 450	0.051 ± 0.018
450 - 500	0.023 ± 0.012
500 - 550	0.074 ± 0.016

Table 2 : Angular anisotropies A = W (0°) / W (90°) - 1

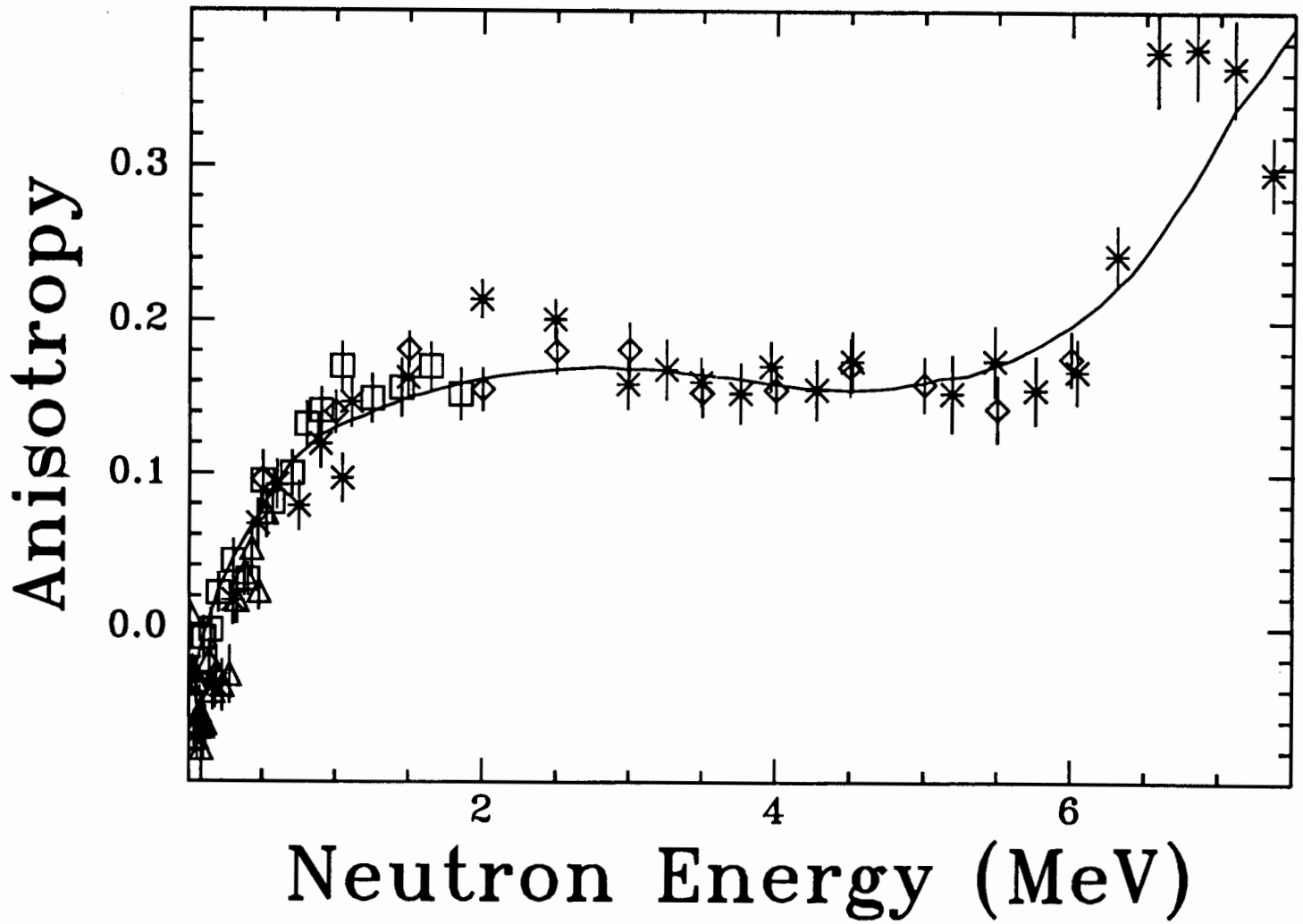

Fig. 1 : Measured fragment anisotropies for 0 - 7.5 MeV. ◇ ref. (4), △ ref. (5), ◻ ref. (9) and ∗ ref. (10).

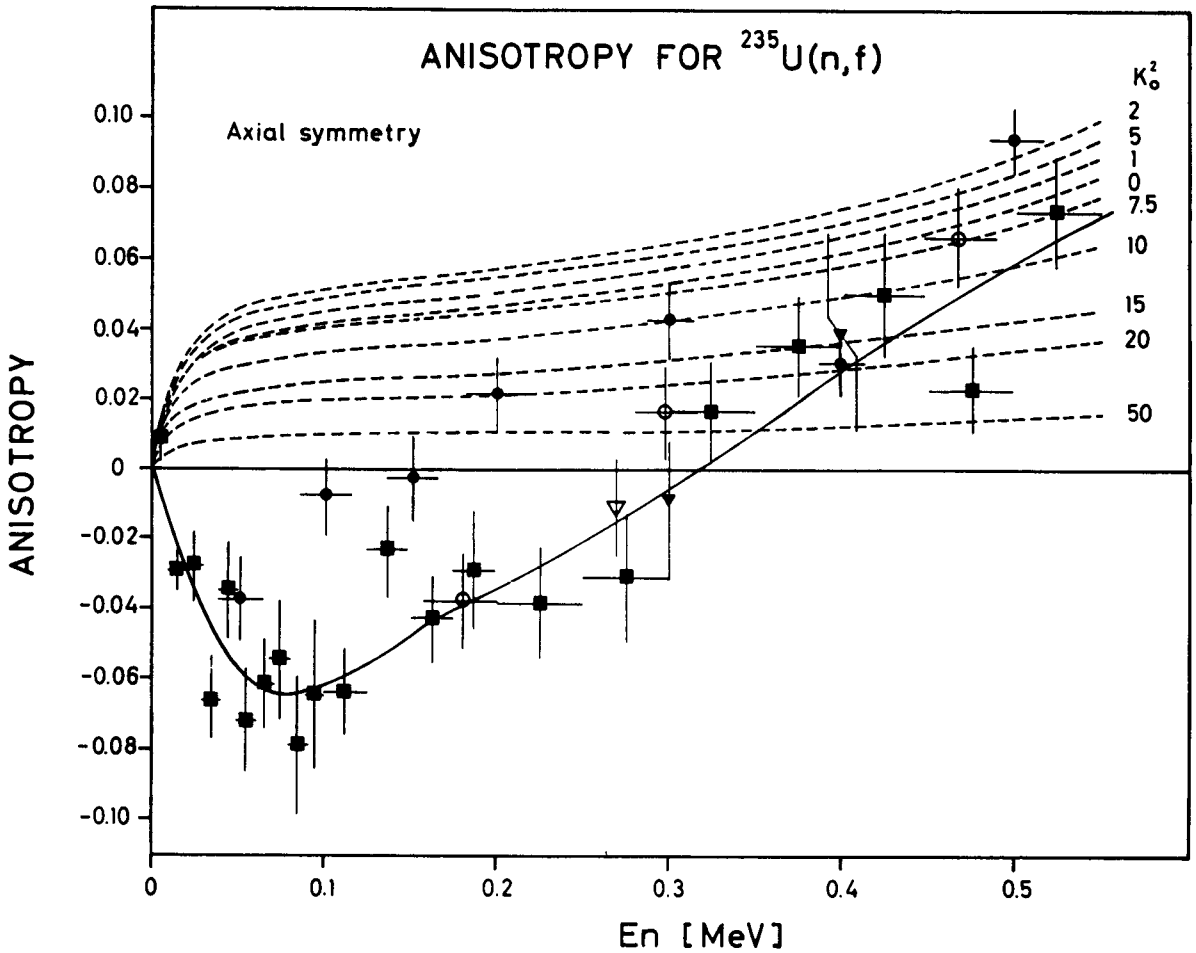

Fig. 2 : Measured fragment anisotropies for 0 - 0.5 MeV. Axial symmetry at the saddle point is assumed. ■ ref. (5), • ref. (9), ○ ref. (10), ▽ ref. (11) and ▼ ref. (12).

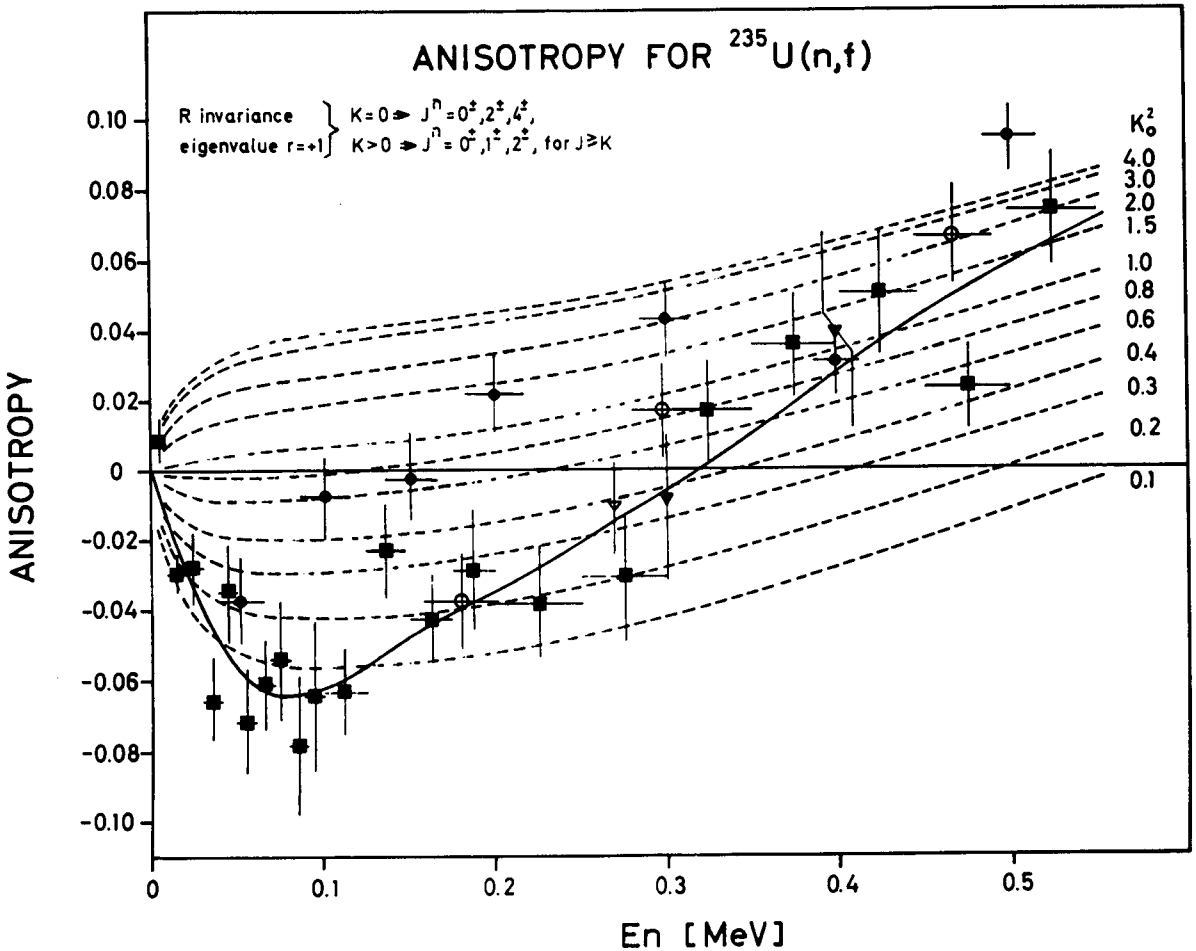

Fig. 3 : Measured fragment anisotropies for 0 - 0.5 MeV. R-invariance at the saddle point is assumed. ■ ref. (5), • ref. (9), ○ ref. (10), ▽ ref. (11) and ▼ ref. (12).

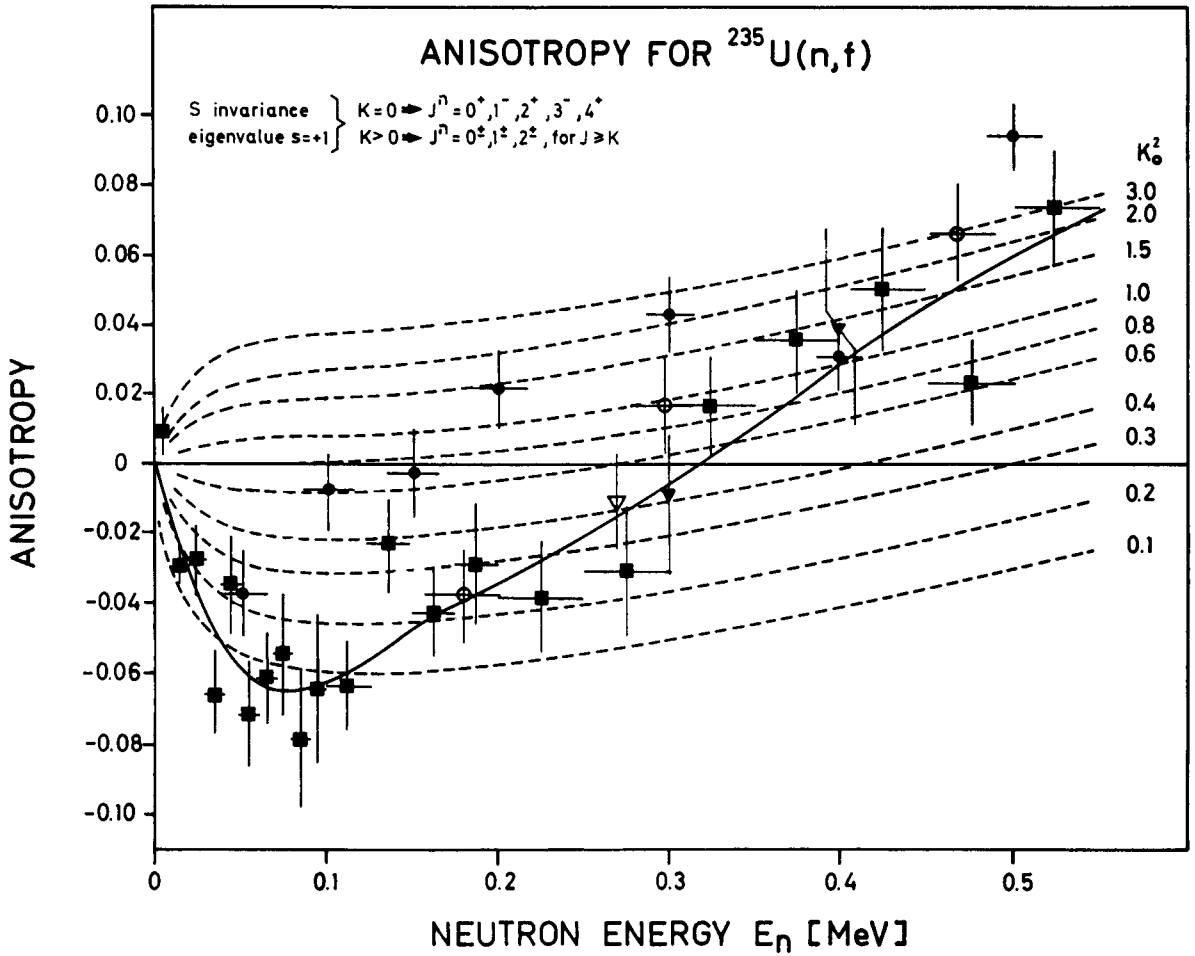

Fig. 4 : Measured fragment anisotropies for 0 - 0.5 MeV. S-invariance at the saddle point is assumed. ■ ref. (5), • ref. (9), ○ ref. (10), ▽ ref. (11) and ▼ ref. (12).

THE ^{238}U FISSION CROSS SECTION

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Introduction

The $^{238}U(n,f)$ cross section is a useful reference standard for neutron cross section measurements in the MeV region for the following reasons: i) the excitation function has a threshold near 1 MeV which eliminates the influence of low-energy neutrons, ii) a smooth variation of the cross section with neutron energy makes it insensitive to energy resolution, iii) a relatively large cross section facilitates measurements of neutron fluxes, iv) a well developed fission-chamber and data-handling-techniques increase the reliability of those measurements, v) the back-to-back fission-chamber techniques allow simultaneous measurements of the $^{238}U(n,f)$ and $^{235}U(n,f)$ cross sections which cross sections are also simultaneously evaluated.

Status

Two early reports by Smith in 1981 (1) and by Kanda in 1984 (2) discussed the $^{238}U(n,f)$ cross section as a reference standard. Recently, two independent evaluations have been performed of the $^{238}U(n,f)$ cross section. One has been made by the Japanese Nuclear Data Committee for JENDL-3 (3) and includes a simultaneous evaluation of some fission and radiative capture cross sections of heavy nuclides and the other made by CSEWG for ENDF/B-VI (4) does include a simultaneous evaluation of the same reactions for heavy nuclides but also a few reactions for light nuclides. The two simultaneous evaluations are independently performed but have used different evaluation procedures. The results are in good agreement.

The major experimental data for the $^{238}U(n,f)$ cross section used in the present evaluations were presented to the meeting on fast fission cross sections held at Argonne National Laboratory in 1976 (5): Some of those data have been published later in journals. The measurements presented after this meeting up to 1984 were reviewed in the references (1,2). A few recent measurements were not discussed in the reviews. They are the ratio measurements to $^{235}U(n,f)$ by Androsenko et al. (6), and by Goverdowskij et al. (7), and the absolute measurement with an associated particle method by Merla et al. (8). However, they will have a marginal effect on the evaluations. The ratio data to the $^{235}U(n,f)$ cross section measured with a d-Be (thick target) neutron source by Watanabe et al. (9) were consistent with the simultaneously evaluated results (10).

Evaluation techniques

The simultaneous evaluation of the $^{238}U(n,f)$ cross sections for JENDL-3 (3) were made utilizing a generalized least squares method and second-order spline functions including the $^{235}U(n,f)$, $^{238}U(n,f)$, $^{239}Pu(n,f)$, $^{240}Pu(n,f)$, $^{241}Pu(n,f)$, $^{197}Au(n,\gamma)$ and $^{238}U(n,\gamma)$ cross sections in the neutron energy range between 50 keV and 20 MeV. In this evaluation, experiments were classified into two groups, absolute and ratio measurements, respectively. The former group included associated particle method experiments and ratio measurements to the H(n,n) cross section. The majority of the latter group was ratio cross section data to $^{235}U(n,f)$ and the remainder ratio data to other cross sections included in the evaluation. Regarding the $^{238}U(n,f)$ evaluation, five experiments were identified as absolute data and eleven as ratio measurements to $^{235}U(n,f)$. Covariances of individual data sets used in the generalized least-squares calculation were estimated on the basis of their experimental conditions described in the reports according to the guide-lines commonly used for every measurements. Correlation between different data sets was not considered.

The ^{238}U fission cross-section is one of the standards which have been evaluated simultaneously for ENDF/B-VI (11,12) by a committee of the U.S. Cross-section Evaluation Working Group consisting of A. Carlson (Chairman), G. Hale, W.P. Poenitz and R. Peelle. In this the cross-sections were determined for $^{6}Li(n,\alpha)$, $^{10}B(n,\alpha_0)$, $^{10}B(n,\alpha_1)$, $^{197}Au(n,\gamma)$ and $^{235}U(n,f)$ along with the important cross-sections $^{238}U(n,\gamma)$, $^{238}U(n,f)$ and $^{239}Pu(n,f)$. The goals were to correctly utilise all types of absolute and ratio data while taking full advantage of the R-matrix representation for the light-element standards. R-matrix analyses of Hale were used for the ^{7}Li and ^{11}B systems, and Poenitz applied his simultaneous least-squares fitting system to the data sets that are independent of those used in the R-matrix fits. The outputs of these studies, including their full co-variance matrices, were combined by Peelle in a single least-squares adjustment. The original full data base was not internally consistent within the quoted uncertainties. The resulting cross-section and errors are listed below for the ^{238}U fission cross-section.

Conclusion

The simultaneous evaluation methods developed in recent years have resulted in highly accurate $^{238}U(n,f)$ cross section on the basis of absolute and ratio measurements. It is consistent with the cross section of $^{235}U(n,f)$ and the other reactions included in the evaluations. The accuracies of the evaluated cross sections do almost meet the requests in WRENDA 83/84. Good agreement is obtained between the two recent evaluations JENDL-3 and ENDF/B-VI.

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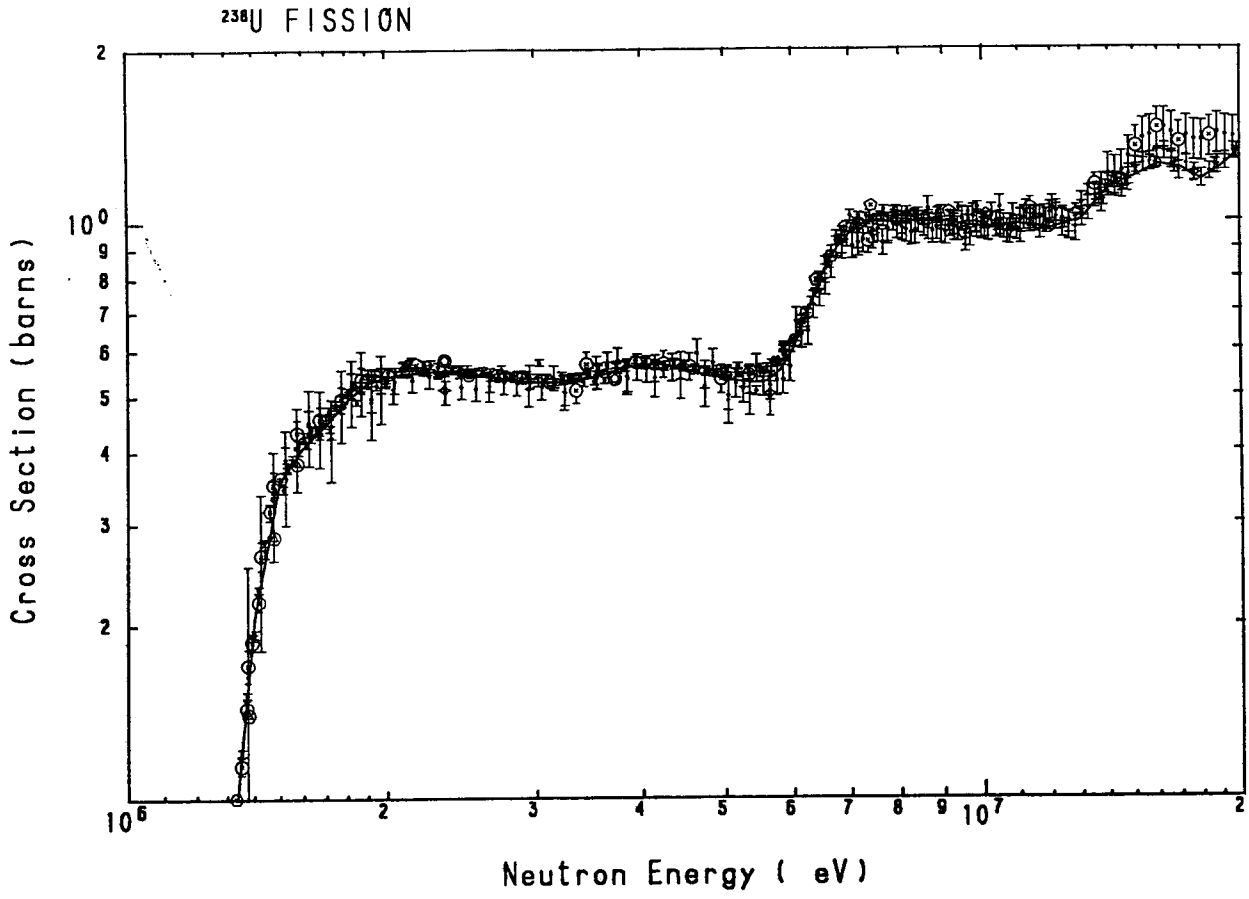

Fig. 1. JENDL-3 evaluated fission cross section of ^{238}U comparing with experiments.

^{238}U FISSION CROSS SECTIONS - Recommended Reference Data

Numerical values from ENDF/B-VI, MAT 9237
Applicable energy range: threshold-20 MeV
Linear - linear interpolation
Uncertainties are taken from the IRDF 90.

Cross section values

E(keV)	XSEC(b)	Uncertainty (percent)	E(keV)	XSEC(b)	Uncertainty (percent)
5.000E02	3.7642E-04	7.57	6.000E02	8.2255E-04	6.36
8.000E02	4.4703E-03	3.93	1.000E03	1.3980E-02	1.50
1.020E03	1.5700E-02	1.48	1.030E03	1.6926E-02	1.47
1.050E03	2.0000E-02	1.45	1.080E03	2.7000E-02	1.43
1.100E03	3.0040E-02	1.41	1.130E03	3.6000E-02	1.38
1.140E03	3.8100E-02	1.37	1.150E03	3.9200E-02	1.36
1.170E03	4.0247E-02	1.34	1.200E03	4.2087E-02	1.31
1.230E03	4.0300E-02	1.28	1.240E03	4.0000E-02	1.28
1.250E03	3.9120E-02	1.27	1.280E03	5.0200E-02	1.24
1.300E03	6.5000E-02	1.22	1.350E03	1.1188E-01	1.17
1.400E03	1.8550E-01	1.13	1.450E03	2.8224E-01	1.08
1.480E03	3.3100E-01	1.05	1,500E03	3.5600E-01	1.03
1.525E03	3.8050E-01	1.01	1.550E03	3.9900E-01	0.98
1.575E03	4.1250E-01	0.96	1.600E03	4.2260E-01	0.94
1.700E03	4.5500E-01	0.84	1.800E03	4.8200E-01	0.75
1.900E03	5.0700E-01	0.75	2.000E03	5.2500E-01	0.72
2.100E03	5.3550E-01	0.73	2.200E03	5.3910E-01	0.73
2.400E03	5.3730E-01	0.75	2.600E03	5.3280E-01	0.73
2.800E03	5.2700E-01	0.79	3.000E03	5.1600E-01	0.75
3.200E03	5.2100E-01	0.75	3.600E03	5.3540E-01	0.78
4.000E03	5.4830E-01	0.81	4.500E03	5.4960E-01	0.82
4.700E03	5.4700E-01	0.89	5.000E03	5.4050E-01	0.87
5.300E03	5.4300E-01	0.92	5.500E03	5.5000E-01	0.90
5.800E03	5.7310E-01	0.95	6.000E03	6.1530E-01	1.03
6.200E03	6.8590E-01	1.04	6.500E-03	8.2570E-01	1.01
6.700E03	8.8700E-01	1.02	7.000E03	9.4030E-01	1.04
7.300E03	9.6800E-01	1.08	7.500E03	9.8070E-01	1.11
7.750E03	9.9100E-01	1.23	8.000E03	9.9350E-01	1.16
8.500E03	1.0000E-00	1.07	9.000E03	9.9790E-01	1.15
1.000E04	9.8680E-01	1.21	1.100E04	9.8300E-01	1.36
1.150E04	9.8300E-01	1.50	1.200E04	9.8500E-01	1.33
1.300E04	1.0130E00	1.04	1.400E04	1.1300E00	0.76
1.450E04	1.1550E00	0.70	1.500E04	1.1980E00	0.94
1.600E04	1.2593E00	1.26	1.700E04	1.2561E00	1.40
1.800E04	1.2493E00	1.44	1.900E04	1.2954E00	1.49
2.000E04	1.3521E00	1.74			

THE $^{27}Al(n,\alpha)\,^{24}Na$ CROSS SECTION

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The $^{27}Al(n,\alpha)^{24}Na$ activation reaction cross section is recommended as a Category-I dosimetry reference cross section. It is widely employed as a standard in dosimetry and activation measurements.

The evaluation of the $^{27}Al(n,\alpha)^{24}Na$ cross section carried out by Tagesen and Vonach in 1981 (1) proved that certain portions of the excitation function are very well known, especially the 14 MeV region. Due to the wealth of consistent data in the 13.40 to 14.90 MeV range the uncertainties of the evaluated cross sections for 0.1 MeV wide energy groups are less than 1.0 %; from 13.6 to 14.8 MeV they amount to 0.3 - 0.6 % only. In the 12.0 - 13.4 MeV and 15 - 20 MeV ranges uncertainties of 2 - 4 % had been obtained, whereas in the 8 - 12 MeV range experimental data were rather scarce; from 8.5 to 9.5 MeV neutron energy the uncertainties of the group cross sections even amounted to $\approx$ 7 - 9 %. Thus, the status of the evaluated cross sections for the $^{27}Al(n,\alpha)^{24}Na$ reaction as a whole had not yet fully met the requirements specified for a category-I dosimetry reaction. Moreover, the existence of structure in the lower-energy region remained an open question.

Since 1981 several precision measurements have been carried out, which cover the entire energy range from near threshold to about 20 MeV, but special attention has been paid to the 5.9 - 13 MeV region. The very valuable information provided by these experiments called for an update of the 1981 evaluation by Tagesen and Vonach. The results of the new evaluation by M Wagner et al (2) performed at the Institut für Radiumforschung, University of Vienna is recommended for use as the present cross section standard for the $^{27}Al(n,\alpha)^{24}Na$ reaction.

The cross sections with uncertainties (1σ) are given in the following table. There are positive correlations up to $\approx$ 60 % between the uncertainties for different neutron energies. The full correlation matrix is given in ref 2.

<u>References</u>

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^{27}Al(n,α) CROSS SECTIONS - Recommended Reference Data

Numerical data from from M Wagner et al, Physics Data, 13-5 (1990)
Linear-linear interpolation

Cross section values

GROUP-ENERGY (MeV) to (MeV)	XSEC (mb)	ERROR (mb)	ERROR (PERCENT)
5.40 - 5.71	0.18	0.02	10.3
5.71 - 6.00	1.08	0.08	7.8
6.00 - 6.20	1.66	0.19	11.6
6.20 - 6.40	3.57	0.15	4.2
6.40 - 6.60	6.28	0.18	2.9
6.60 - 6.80	9.48	0.26	2.7
6.80 - 7.00	14.52	0.47	3.2
7.00 - 7.25	20.39	0.55	2.7
7.25 - 7.50	24.06	0.63	2.6
7.50 - 7.75	31.11	0.78	2.5
7.75 - 8.00	39.21	0.97	2.5
8.00 - 8.25	43.09	1.00	2.3
8.25 - 8.50	50.35	0.93	1.8
8.50 - 8.75	61.48	1.26	2.1
8.75 - 9.00	67.14	1.11	1.7
9.00 - 9.50	78.29	1.58	2.0
9.50 - 10.00	86.67	1.49	1.7
10.00 - 10.50	91.39	2.78	3.0
10.50 - 11.00	103.53	2.90	2.8
11.00 - 11.50	111.43	3.54	3.2
11.50 - 12.00	115.28	2.54	2.2
12.00 - 12.40	120.36	2.24	1.9
12.40 - 12.80	123.90	2.48	2.0
12.80 - 13.00	125.40	3.10	2.5
13.00 - 13.20	124.18	2.56	2.1
13.20 - 13.40	127.00	2.26	1.8
13.40 - 13.55	125.67	0.78	0.6
13.55 - 13.65	125.30	0.66	0.5
13.65 - 13.75	124.45	0.60	0.5
13.75 - 13.85	123.23	0.63	0.5
13.85 - 13.95	122.32	0.64	0.5
13.95 - 14.05	122.84	0.46	0.4
14.05 - 14.15	121.57	0.57	0.5
14.15 - 14.25	121.45	0.66	0.5
14.25 - 14.35	119.40	0.67	0.6
14.35 - 14.45	116.76	0.33	0.3
14.45 - 14.55	115.53	0.60	0.5
14.55 - 14.65	114.11	0.55	0.5
14.65 - 14.75	112.49	0.46	0.4

GROUP-ENERGY (MeV) to (MeV)	XSEC (mb)	ERROR (mb)	(PERCENT)
14.75 - 14.85	111.90	0.44	0.4
14.85 - 14.95	110.14	0.93	0.8
14.95 - 15.50	107.19	1.50	1.4
15.50 - 16.00	98.35	1.72	1.7
16.00 - 16.50	90.73	1.97	2.2
16.50 - 17.00	81.39	1.67	2.1
17.00 - 18.00	67.21	1.36	2.0
18.00 - 19.00	56.94	1.36	2.4
19.00 - 20.00	43.15	1.37	3.2
20.00 - 21.00	33.87	1.74	5.1

THE ^{59}Co (n,2n) ^{58}Co CROSS SECTION

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The NEANDC Subcommittee recommends the use of the reactions $^{93}Nb(n,2n)^{92m}Nb$ and $^{59}Co(n,2n)^{58}Co$ as secondary standards for flux measurements in the neutron energy range around 14 MeV. These reactions have very convenient half-lives (10 days and 71 days, respectively) to serve as neutron flux monitors for long irradiations and their decay properties are very well known and suited for accurate absolute activity measurements.

It is recommended to use the cross section data from the evaluation of M Wagner et al (1) performed at the Institut für Radiumforschung, University of Vienna.

The cross sections and their uncertainties (1σ) are given in the following table. There are positive correlations up to ≈ 50 % between the uncertainties for different neutron energies. The full correlation matrix is given in ref 1.

<u>Reference</u>

1. M Wagner, H Vonach, A Pavlik, B Strohmaier, S Tagesen, and J Martinez-Rico, Pysics Data, 13-5, Fachinformationszentrum, Karlsruhe (1990)

$^{59}Co(n,2n)^{58}Co$ CROSS SECTIONS - Recommended Reference Data

Numerical data from from M Wagner et al, Physics Data, 13-5 (1990)
Linear-linear interpolation

Cross section values

GROUP-ENERGY (MeV) to (MeV)	XSEC (mb)	ERROR (mb)	ERROR (PERCENT)
10.80 - 11.00	16.64	6.91	41.5
11.00 - 11.50	87.72	8.04	9.2
11.50 - 12.00	202.28	15.54	7.7
12.00 - 12.50	348.28	23.07	6.6
12.50 - 13.00	490.98	16.62	3.4
13.00 - 13.50	575.15	7.04	1.2
13.50 - 14.00	662.86	6.14	0.9
14.00 - 14.25	704.13	14.82	2.1
14.25 - 14.50	750.74	18.72	2.5
14.50 - 14.75	769.72	6.50	0.8
14.75 - 15.00	788.10	7.21	0.9
15.00 - 16.00	794.00	30.97	3.9
16.00 - 17.00	809.49	30.29	3.7
17.00 - 18.00	835.09	38.24	4.6
18.00 - 19.00	851.85	22.10	2.6
19.00 - 20.00	876.23	29.85	3.4
20.00 - 21.00	848.29	47.46	5.6
21.00 - 22.00	734.57	46.60	6.3
22.00 - 23.00	629.63	63.04	10.0

THE ^{93}Nb (n,2n) ^{92m}Nb CROSS SECTION

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The NEANDC Subcommittee recommends the use of the reactions ^{93}Nb(n,2n)^{92m}Nb and ^{59}Co(n,2n)^{58}Co as secondary standards for flux measurements in the neutron energy range around 14 MeV. These reactions have very convenient half-lives (10 days and 71 days, respectively) to serve as neutron flux monitors for long irradiations and their decay properties are very well known and suited for accurate absolute activity measurements.

It is recommended to use the cross section data from the evaluation of M Wagner (1) performed at the Institut für Radiumforschung, University of Vienna.

The cross sections and their uncertainties (1σ) are given in the following table. There are positive correlations up to ≈ 70 % between the uncertainties for different neutron energies. The full correlation matrix is given in ref 1.

<u>Reference</u>

1. M Wagner, Report INDC(AUS)-D14, Vienna, Oct 1991

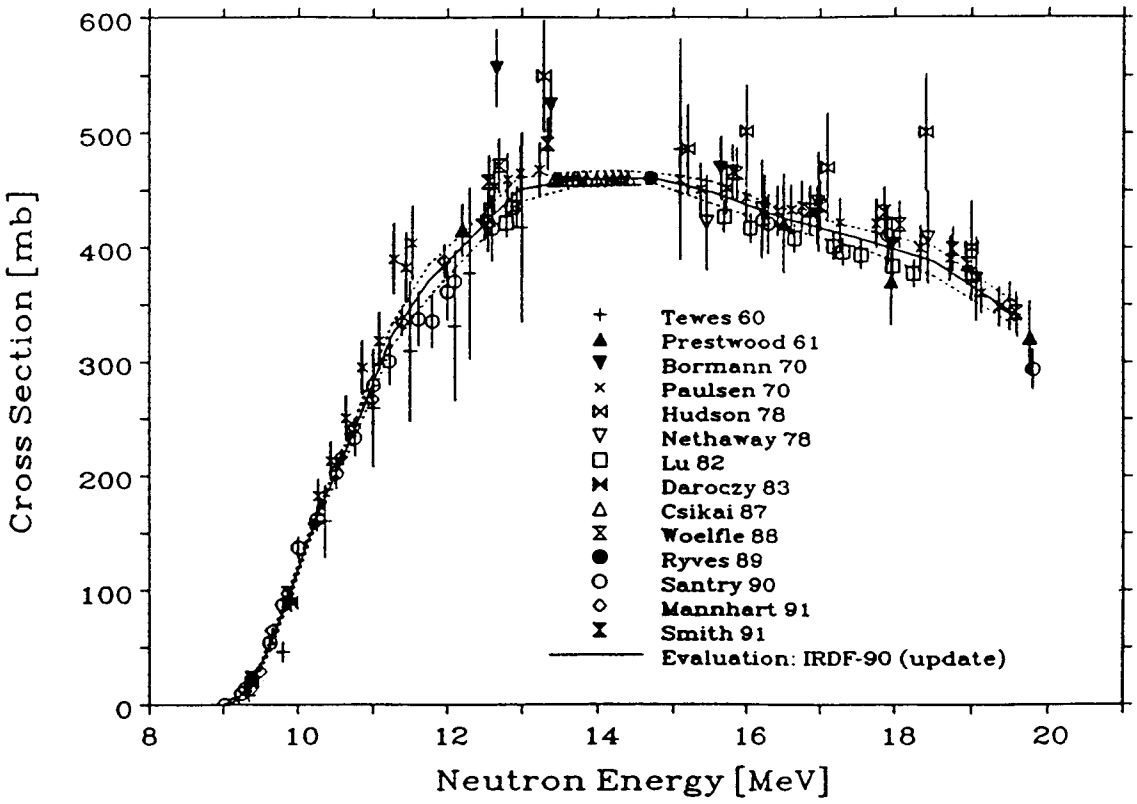

Figure 1 Experimental cross sections (renormalized and corrected) for the reaction $^{93}Nb(n,2n)^{92m}Nb$ together with the evaluated cross sections

^{93}Nb(n,2n)^{92m}Nb CROSS SECTIONS - Recommended Reference Data

Numerical data from from M Wagner, Report INDC(AUS)-D14, Vienna, Oct 1991
Linear-linear interpolation

Cross section values

GROUP-ENERGY (MeV) to (MeV)	XSEC (mb)	ERROR (mb)	ERROR (PERCENT)
9.00 - 9.25	0.99	1.02	102.6
9.25 - 9.50	15.94	2.01	12.6
9.50 - 9.75	51.24	5.00	9.8
9.75 - 10.00	95.32	5.79	6.1
10.00 - 10.50	164.49	5.57	3.4
10.50 - 11.00	241.48	8.08	3.3
11.00 - 11.50	322.79	9.90	3.1
11.50 - 12.00	370.53	14.25	3.8
12.00 - 12.50	403.52	11.71	2.9
12.50 - 13.40	449.24	10.59	2.4
13.40 - 14.00	458.65	6.79	1.5
14.00 - 14.50	459.82	6.79	1.5
14.50 - 15.00	459.54	5.00	1.1
15.00 - 16.00	447.50	9.49	2.1
16.00 - 17.00	424.86	8.33	2.0
17.00 - 18.00	408.27	8.16	2.0
18.00 - 19.00	387.00	12.90	3.3
19.00 - 20.00	344.16	10.33	3.0

NEUTRON ENERGY STANDARDS

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A common scale of neutron energies is generally required for comparison and cross-correlation of different experimental results. The accuracy may be particularly demanding when comparing different reaction channels, where the measured amplitudes may be widely different, and the identification of the excited levels is based only on their energy.

Establishing an absolute energy scale of a neutron spectrometer often implies a lengthy and complicated experimental effort; this may be avoided with the help of suitable energy standards.

All neutron energy standards are at present based on time-of-flight measurements.

In this type of spectrometers, the flight-path length is in general the worst known parameter. Clocks used to measure flight times generally have adequate accuracy and do not have calibration problems. However, moderation of source neutrons and scattering in the detector may introduce energy-dependent delay times that need careful evaluation.

The table of standard energies.

The data listed below are based on the recommended neutron energy standards (1) published by IAEA in 1983 as part of the INDC-NEANDC nuclear standards file.

In updating the standard table, account was taken of new information as follows.

1) The energy dependent moderation distances calculated for ORELA (2) were used to correct the data of refs. (3) and (4). This led to a general increase in the energy values, especially at high energies.

2) After the above corrections, the ^{238}U data resulting from measurements carried out at Oak Ridge (3), Harwell (4) and Geel (5) showed the existence of slight systematic differences in the energy scales. In order to obtain a consistent set of data, an average energy scale was adopted to normalize all data below 1 MeV, including the low-energy values of ref.(7) as well as those reported in refs. (4) and (8). The standard values reported in ref. (9) are averages of such normalized data.

New measurements of flight-path lengths carried out at ORELA (10) showed that old data based on length measurements performed with surveying tapes, might be affected by much larger errors than quoted in refs (3) and (4). As a consequence, in ref. (9) the errors due to this uncertainty were increased.

3) In 1990, precise values of ^{238}U resonance energies were obtained at Oak Ridge (11), between 1 and 6 keV. These data were used to re-normalize the energy scale of ^{238}U. Below 1.4 keV, this led to a decrease in the energies, almost exactly back to the original values of ref. (3). The errors quoted in ref. (9) could be somewhat reduced.

4) The last atomic mass evaluation shows that the neutron mass is appreciably higher ($\approx$ 15 ppm) than the values used at Harwell (5) and Oak Ridge (3) to convert the measured neutron velocity into energy. The corresponding energies were increased according to the latest adjustment of the fundamental physical constants (12).

5) Above 0.2 MeV, the data accuracy on ^{56}Fe and ^{32}S could be remarkably improved on account of measurements performed at Oak Ridge (13) and Geel (14), respectively.

6) Above 3 MeV, where the reference data were scarce, five new resonances were added (one is a closely spaced doublet that may be used to check the resolution). These data were taken from very precise and well documented results obtained at Karlsruhe (15).

Above 10 MeV the only available calibration point is given by a broad peak in the cross-section of ^{12}C; the value given in the standard table is an unpublished result obtained at Karlsruhe, already reported in ref. (8).

Between 1 and 10 MeV the first two values, 1652.1 keV of ^{16}O and 2078.6 keV of ^{12}C, result from data obtained at Karlsruhe (16) and Geel (17), and at Karlsruhe (16), Geel (17) and Harwell (5), respectively. All other data at higher energies are taken from measurements performed at the Karlsruhe isochronous cyclotron (15).

More details are available in ref. (9)

Recommendations

Below 1 eV there is still a lack of standard values. The cross-section dip corresponding to the first oscillator level at 0.137 eV for protons in ZrH may provide a useful standard (18), but the numerical value and error are not yet available.

An experimental effort is also needed to establish some reference points below 0.1 eV; these could be provided by cross-section edges of crystalline materials.

In the region above 1 MeV it is suggested by Wiedling (19) that some reaction thresholds will be adopted as standards that could find application in the calibration of monoenergetic neutron generators, and also in time-of-flight systems.

It must be pointed out that part of the presently available standard values are based on works lacking a satisfactory description of the adopted procedures. It is recommended that reports on high-precision energy determinations strive to avoid such shortcomings.

Since neutron energy standards may reach accuracies below 20 ppm, it seems advisable to recommend that all energy calibrations consistently use the same value of the neutron mass.

The recommended value is

$$M_n = 939.56563 \text{ MeV}$$

This value is quoted by Cohen and Taylor (12) with an error of 0.3 ppm in their 1986 adjustment of fundamental physical constants.

Using the standard value of the velocity of light, the conversion constant from velocity to energy becomes

$$M_n/2c^2 = 5227.039 \text{ [MeV (m/ns)}^{-2}\text{]}$$

also with an error of 0.3 ppm.

For the same reason, in time-of-flight calibrations, it is recommended that a relativistic correction will be applied starting from quite low energies. For instance, it should be kept in mind that the classical formula deviates from the relativistic one by more than 20 ppm above 12.5 keV.

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NEUTRON ENERGY STANDARDS - Recommended Reference Data

Peak energies of neutron resonances to be used as energy standards (laboratory system).

Isotope	E(eV)	Error(eV)	Isotope	E(eV)	Error(eV)
Ir-191	6.529E-01	± 5.0E-04	U-238	4.51231E03	± 4.5E-01
Ir-193	1.298E00	± 1.0E-03	U-238	5.65054E03	± 5.7E-01
U-238	6.673E00	± 1.0E-03	S-32	3.0388E04	± 2.3E01
U-238	2.0864E01	± 3.0E-03	Na-23	5.321E04	± 4.0E01
U-238	3.6671E01	± 6.0E-03	Pb-206	7.122E04	± 5.0E01
U-238	6.6015E01	± 1.0E-02	S-32	9.755E04	± 7.0E01
U-238	8.0729E01	± 1.2E-02	S-32	1.1223E05	± 8.0E01
U-238	1.4563E02	± 2.0E-02	Fe-56	2.67044E05	± 2.2E01
U-238	1.8964E02	± 3.0E-02	S-32	4.1268E05	± 7.0E01
U-238	3.1128E02	± 5.0E-02	S-32	8.1934E05	± 1.8E02
U-238	3.9759E02	± 6.0E-02	O-16	1.6521E06	± 6.0E02
U-238	4.6315E02	± 7.0E-02	C-12	2.0786E06	± 5.0E02
U-238	6.1997E02	± 9.0E-02	O-16	3.21165E06	± 4.0E01
U-238	7.0828E02	± 1.1E-01	O-16	3.43857E06	± 6.0E01
U-238	9.0504E02	± 1.4E-01	O-16	3.44160E06	± 4.0E01
U-238	1.41989E03	± 1.4E-01	O-16	4.59479E06	± 7.0E01
U-238	1.47386E03	± 1.5E-01	C-12	4.93703E06	± 7.0E01
U-238	2.48915E03	± 2.5E-01	O-16	5.36947E06	± 1.1E02
U-238	2.67211E03	± 2.7E-01	O-16	6.07614E06	± 1.1E02
Pb-206	3.3574E03	± 4.0E-01	C-12	6.29677E06	± 3.9E02
U-238	3.45796E03	± 3.5E-01	C-12	1.2087E07	± 9.0E03

STATUS OF ACTINIDE HALF LIVES

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The recommended reference half-life data for the major actinides, included in the 1985 revision of the 1983 INDC/NEANDC Standards File, resulted from an extensive international review of transactinium nuclide decay data performed by an IAEA Coordinated Research Programme (CRP) from 1978 to 1985. The results of this effort were published in the IAEA Technical Report Series No. 261 (1986). The complete listings, published in this IAEA report, include decay data for a wide range of heavy elements of broader interest than that of nuclear standards.

At the end of 1989, a Specialist's Meeting on the Status and the Requirements of Transactinium Isotope Decay Data reviewed the data in the light of new measurements and/or evaluations. In a number of cases, data have been supplemented or replaced by values measured or evaluated by members of the CRP and by other groups or individuals.

The anticipated updating of the IAEA Technical Report 261 on Decay Data of Transactinium Nuclides is delayed. Among the half-life values, specifically ^{241}Pu requires updating, in view of recent experiments. A new evaluation may produce a value around 14.35 years compared to the presently recommended value of 14.4 ± 0.1 years.

ACTINIDE HALF-LIVES - Recommended Reference Data

Nuclide	Decay mode	Half-life and Uncertainty Years
U-233	Alpha	(1.592 ± 0.002) E05
	Spont. fission	> 2.7 E17
U-234	Alpha	(2.457 ± 0.003) E05
	Spont. fission	(1.42 ± 0.08) E16
U-235	Alpha	(7.037 ±0.007) E08
	Spont. fission	(1.0 ± 0.3) E19
U-238	Alpha	(4.47 ± 0.02) E09
	Spont. fission	(8.2 ± 0.1) E15
Np-237	Alpha	(2.14 ± 0.01) E06
	Spont. fission	> 1. E18
Pu-239	Alpha	(2.411 ± 0.003) E04
	Spont. fission	(8. ± 2.) E15
Pu-240	Alpha	(6.563 ± 0.007) E03
	Spont. fission	(1.16 ±0.02) E11
Pu-241	Alpha	(5.96 ± 0.04) E05
	Beta	(1.44 ± 0.01) E01
Pu-242	Alpha	(3.75 ± 0.02) E05
	Spont fission	(6.77 ± 0.07) E10
Pu-244	Alpha	(8.00 ± 0.09) E07
	Spont. fission	(6.6 ± 0.2) E10
Cf-252	Alpha	(2.73 ± 0.01) E00
	Spont. fission	(8.55 ± 0.03) E01
	Total	(2.645 ± 0.008) E00

THERMAL PARAMETERS FOR ^{233}U, ^{235}U, ^{239}Pu, ^{241}Pu

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The thermal cross sections of ^{235}U are considered as standard reference data for cross section measurements of other nuclides. The thermal neutron data of ^{233}U, ^{235}U, ^{239}Pu, and ^{241}Pu are correlated, because cross section ratios between these nuclides have been measured in addition to some accurate absolute values. The values for 2200 m/s neutrons (0.0253 eV) are used for normalization of cross section curves at thermal and higher energies.

The following table shows a comparison of the recommended values of ENDF/B-VI and the ones adopted in the recent file JEF-2

		ENDF/B-VI	JEF-2
	σ_f	584.25 ± 1.11	582.5
^{235}U	σ_g	98.25 ± 0.74	98.8
	ν_t	2.4320 ± 0.0036	2.437
	σ_f	531.14 ± 1.33	528.45
^{233}U	σ_g	45.51 ± 0.23	45.76
	ν_t	2.4946 ± 0.0040	2.4947
	σ_f	747.99 ± 1.37	747.2
^{239}Pu	σ_g	271.43 ± 2.14	270.2
	ν_t	2.8815 ± 0.0052	2.877
	σ_f	1012.68 ± 6.58	1011.88
^{241}Pu	σ_g	361.29 ± 4.95	362.95
	ν_t	2.9453 ± 0.0059	2.932
^{252}Cf	ν_t	3.7676 ± 0.0049	

The ENDF/B-VI values are based on Axton's evaluation. The JEF-2 values are the result of an important benchmarking of integral data (mainly k_{eff} measurements for a high number of thermal neutron critical experiments). That is why we can observe little differences between the ENDF/B-VI values and the JEF-2 ones.

For the nu-bar ratio data the insufficient knowledge of the fission neutron spectra of the fissile nuclides used to be a significant source of uncertainties. For ^{252}Cf the fission neutron spectrum shape is now better known from Mannhart's evaluation. It would now be essential to establish reliable spectrum shapes also for the other fissile nuclides. Thereafter one should investigate whether the improved spectrum shapes have a noticeable impact on the corrections for the nu-bar ratio experiments. As the recommended nu-bar values have errors of 0.15 to 0.2 percent, even small changes in nu-bar ratio corrections may have an impact on the thermal parameters.

LOW ENERGY CROSS-SECTIONS OF THE MAJOR U AND Pu ISOTOPES

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Introduction

The low energy cross-sections (E_n < 1 eV) are very important for the calculation of the Westcott g-factors, since these are determined by $\sigma(E)$ in the neutron energy range of the corresponding Maxwellian spectrum (about 1 meV to 1 eV) at room temperature.

Furthermore, a good knowledge of $\sigma(E)$ of various U and Pu isotopes is required for a precise understanding of core neutronics in thermal reactor systems.

Status report

Since the publication of the preceding report on nuclear data standards (1), an unexpectedly large number of cross-section measurements has been performed in the subthermal neutron energy region (2-11). Most of them were triggered by requests of reactor physicists (12-14) and tried to fulfill stringent accuracy requirements. For the neutron energy dependence (shape) of η and σ_f for ^{235}U and σ_g for ^{238}U e.g., an accuracy of ± 0.5% was needed in the range from 3 meV to 1 eV to meet the reactor physicists' needs (12, 13).

Measurements on these three quantities have been performed at the Geel Linear Accelerator (3,5,7-10), which has been equipped for this purpose with a liquid nitrogen (77 ^{o}K) cooled methane moderator, yielding about five times more neutrons below 20 meV compared to the usual water-beryllium moderator at room temperature.

(i) Wagemans et al. (3) used this set-up to measure $\sigma_f(E)$ for ^{235}U down to 2 meV. As shown in fig. 1, their experimental data above thermal energy agree with ENDF/B-V within the experimental errors. Below this energy, an agreement exists within two standard deviations, although the measured data below 10 meV clearly go faster to a 1/v-shape than the evaluated curve. The same tendency is present in the σ_f -data of Deruytter et al. (15) and Gwin et al. (16). Thus, the most recent measurements qualitatively support the flatter $\sigma_f(E)\sqrt{E}$-shape proposed by Santamarina et al.(12).

(ii) Corvi and Fioni (5) measured the shape of $\sigma_g(E)$ for ^{238}U in the neutron energy range from 2 meV up to 1 eV. As shown in fig. 2, their results confirm the approximate 1/v dependence foreseen by most evaluations, hence contradicting the shape proposed by Santamarina et al. (12).

(iii) Weigmann et al. (8) performed $\eta(E)$-measurements for ^{235}U at the Geel Linear Accelerator and also at a neutron guide of the High Flux Reactor of the ILL (Grenoble) equipped with two choppers. After corrections for background, scattering of slow and fission neutrons and self-absorption of the γ-rays in the black capture sample, the results of both experiments were in agreement, showing a slight decrease of η in the subthermal region.

These data were confirmed by a recent experiment of Weigmann et al (10) at the Geel Linear Accelerator, who measured the capture-to-fission ratio a for ^{235}U. Assuming a constant average number of neutrons per fission $\nu = 2.4251$, the η-curve shown in fig 3 was obtained.

Moxon et al. (6), on the other hand, measured $\eta(E)$ for ^{235}U in the neutron energy range from 3 - 400 meV, using the Condensed Matter Target of the Harwell Linear Accelerator. The large uncertainties on the data below 20 meV do not allow any conclusion on the shape of η in this crucial energy region.

To improve this situation, Moxon et al (11) performed a new experiment at the Oak Ridge Linear Accelerator. Preliminary results also indicate a slight decrease of η in the subthermal region.

In any case, the results of Weigmann et al (8,10) as well as those of Moxon et al (6,11) clearly contradict the significant increase of η in the neutron energy range from 60 to 120 meV suggested by Halsall (13).

Another important request of the reactor physicists concerned the total cross-section of ^{240}Pu, for which an accuracy < 1% was desired (2,14). This accuracy was achieved down to 6 meV neutron energies by Spencer et al. (2) at the Oak Ridge Linear Accelerator. Their σ_t (E) data below 0.1 eV are about 2 % lower than the corresponding ENDF/B-V curve, as can be seen in fig. 4.

A last group of $\sigma(E)$ measurements were made with the purpose of obtaining more accurate values of the Westcott g_f - factors or of the 2200 m/s total cross-sections.

Wagemans et al. (3,7,9) measured σ_f (E) for ^{233}U, ^{235}U , ^{239}Pu and ^{241}Pu at the Geel Linear Accelerator in the neutron energy region from 2 meV up to 20 eV.

Wagemans et al. (4) also determined the shape of the $^{235}U(n,f)$ cross-section with very cold neutrons (6-60 meV). These measurements were performed at

guide. Within the precision, these results do not show a significant deviation of the $^{235}U(n,f)$ cross-section from a 1/v-shape. This confirms that the extrapolated values of σ_f in the neutron energy region from zero to 2 meV need to be very close to a 1/v extrapolation.

Table 1 compares the values of the Westcott g-factor calculated from the data of Wagemans et al. (3,7,9) with the ENDF/B-V and -VI evaluated values. The relatively high uncertainty on g_f for ^{241}Pu is a reflection of the rather poor σ_f - data base < 1 eV available for this nucleus.

Finally, Spencer et al. (2) measured σ_t (E) for ^{235}U and ^{239}Pu at the Oak Ridge Linear Accelerator down to 6 meV. New 2200 m/s σ_t -values were deduced from these experiments.

In the mean time, most of these new experimental data have been taken into account in the latest JEF-2 evaluation by Tellier (17). An independent evaluation of the low-energy cross sections of ^{235}U was performed by de Saussure et al (18).

Conclusions

The new $\sigma(E)$ or $\eta(E)$-data reported in this review considerably improve the data base < 1 eV and allow a partial check of shapes suggested by reactor physicists. All these recent measurements were performed at pulsed neutron beams, hence all $\sigma(E)$ data points were recorded simultaneously and under the same experimental conditions. This was not the case for a large fraction of the older $\sigma(E)$-data in the meV-region, which were measured point by point at neutron spectrometers. In this way high neutron fluxes can be realized, but the method is subject to solid state effects and higher order diffraction. Anyhow, in all cases the determination of the background components in the subthermal energy region remains a problem which needs great care and attention.

This background problem appears to be extremely difficult in the case of the $\eta(E)$-measurements, in such a way that it considerably hampers the realization of a 0.5% accuracy on the shape of $\eta(E)$. It is the reviewer's conviction that a reliable shape of $\eta(E)$ for ^{235}U with a trustworthy 0.5 % accuracy can only be achieved if the results of several independent experiments coincide. Hence a continuation of the activities of Moxon et al. and Weigmann et al. is strongly endorsed and a critical comparison of their results is recommended.

As far as the g_f -factor is concerned, the new σ_f (E)-data in the neutron energy region 0.001-1 eV allow a more accurate calculation of g_f for ^{241}Pu.

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Table 1: Westcott g_f -factors at T = 20.44 ^{o}C.

Reference	^{233}U	^{235}U	^{239}Pu	^{241}Pu
ENDF/B-V	0.9966	0.9775	1.0582	1.0452
ENDF/B-VI	0.9955±0.0014	0.9771±0.0008	1.0563±0.0021	1.0450±0.0053
Wagemans[a)]	0.994±0.003	0.976±0.002	1.055±0.003	1.041±0.003

a) these g_f -values were calculated only based on the σ_f (E) data of ref. 3,7,9, which are not included in the ENDF/B evaluations.

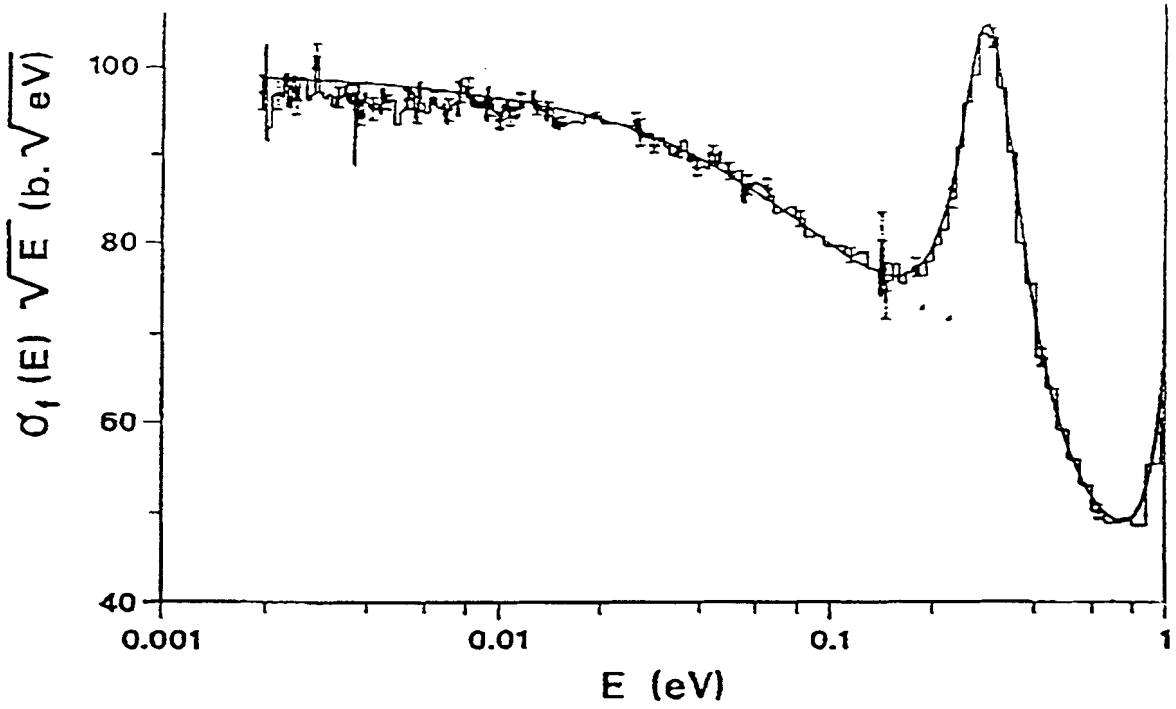

Fig. 1: $\sigma_f(E)\sqrt{E}$ for ^{235}U. The histogram represents the measured data of Wagemans et al. (3), the full line is the ENDF-B5 curve renormalized to σ_f° = 584.25 b.

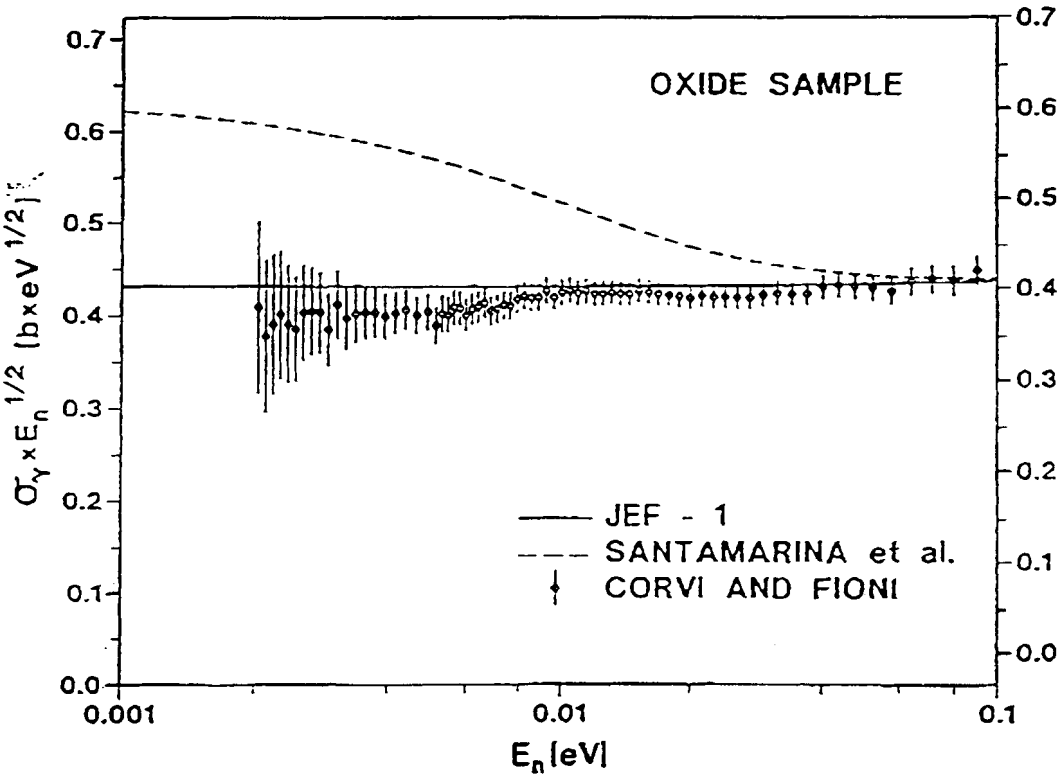

Fig. 2: $\sigma_\gamma(E)\sqrt{E}$ for ^{236}U. The points represent the experimental data of Corvi and Fioni (5). The full line is the JEF-1 evaluation and the intermittent line is the shape proposed by Santamarina et al. (12).

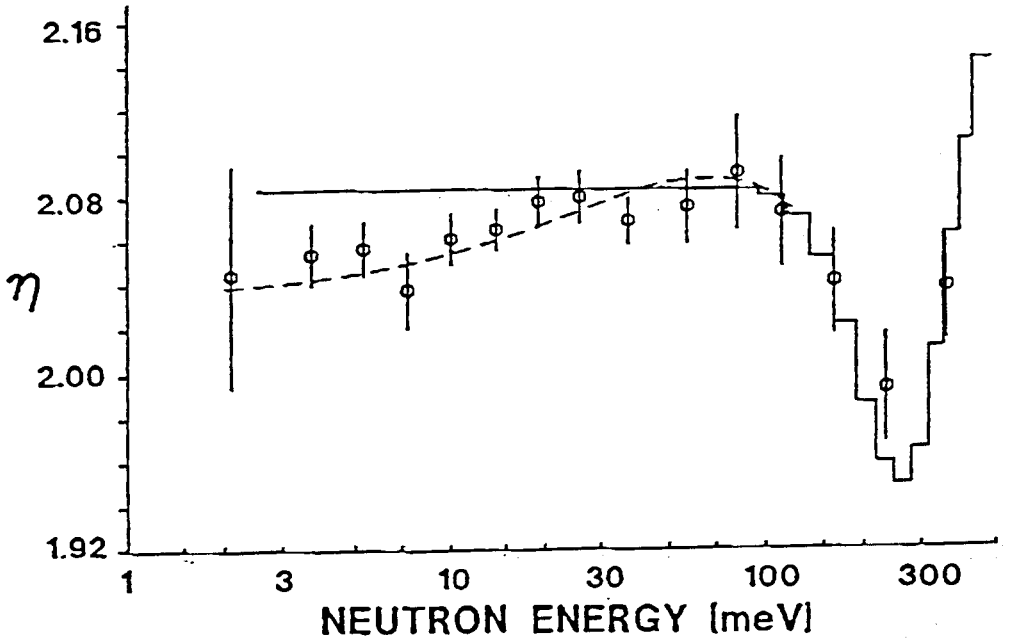

Fig. 3 : $\eta(E)$ for ^{235}U. The full line is the ENDF-B5 evaluation and the intermittent line the-shape proposed by Santamarina et al. (12). The open circles are the experimental data of Weigmann et al. (10).

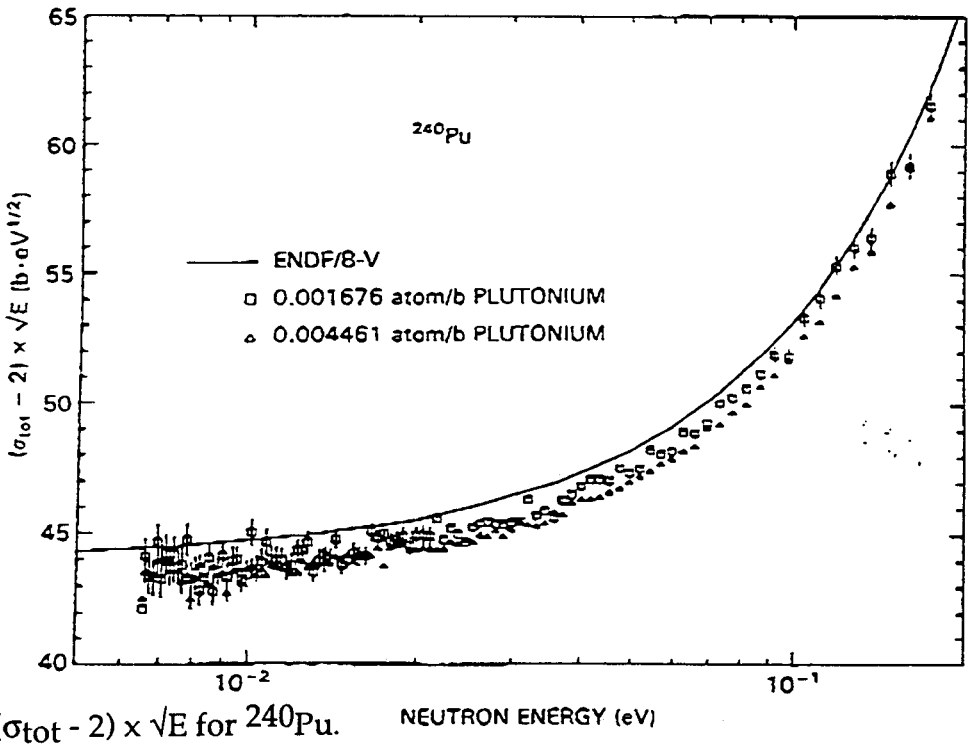

Fig. 4: $(\sigma_{tot} - 2) \times \sqrt{E}$ for ^{240}Pu.
The squares and triangles are the data of Spencer et al. (2). The full line is the ENDF/B-V evaluation. A constant 2b scattering was assumed

THE FISSION NEUTRON SPECTRUM OF ^{252}Cf

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The fission neutron spectrum of ^{252}Cf is used as a reference in measurements of fission neutron spectra of other nuclei, the calibration of neutron detectors, and the testing of dosimetry cross sections. It also impacts on the interpretation of ^{252}Cf ν measurements.

Status

Experimental Differential Data

About forty measurements of the neutron spectrum of the spontaneously fissioning ^{252}Cf have been made during the last thirty years. The measurements carried out until 1979 have been discussed in a review by Blinov (1) which is recommended for consideration and reference of the earlier experiments. The more recent measurements are summarized in Table 1. Several of these experiments were presented and discussed at conferences and specialist's meetings (2-6). Excellent reviews of the more recent experiments (7-19) as well as the newer theoretical models has been given by Maerten and Seeliger (20) and by Boldeman (21).

Table 1. Recent Measurements of the ^{252}Cf Fission Neutron Spectrum.

Year	Reference	Energy Range, MeV	Method
1979-85	Boldeman et al. (15)	1.0-14.3	TOF, plastic scintillator
		0.124-2.66	TOF, Li glass scintillator
1979-82	Blinov et al. (7)	0.003-1.0	TOF, LiH scintillator
1981	Bench and Jasicek (8)	0.9-10.0	Proton recoil, gas counters
"	Mon Jiang-shen et al. (9)	0.45-15.0	TOF, liquid scintillator
"	Bolshov et al. (10)	1.0-11.0	TOF, liquid scintillator
"	Starostov et al. (16)	0.01-10.0	TOF, ^{235}U fission chamber
1982-83	Maerten et al. (11)	9.0-30.2	TOF, liquid scintillator
" "	Poenitz and Tamura (12)	0.2-10.0	TOF, black neutron detector, PSD
1982-86	Boettger et al. (13)	2.0-14.0	TOF, liquid scintillator, PSD
1983	Laytai et al. (14)	0.025-1.18	TOF, Li glass scintillator
1983-85	Blinov et al. (17)	0.04-11.4	TOF, ^{235}U fission chamber
" "	Boettger et al. (18)	5.0-28.0	TOF, liquid scintillator, PSD
1984	Boytsov and Starostov (16)	0.01-3.0	TOF, ^{235}U fission chamber, antracene
1986	Chalupka et al. (19)	14.0-28.0	TOF, liquid scintillator, PSD

Many problems associated with the measurements of this spectrum have been recognized only recently and have been taken care of in some of the later experiments. Consequently, the newer data are in much better agreement (± 3-20%) compared with the older data (factor of 2). However, discrepancies - i.e. differences outside the quoted uncertainties - persist among the newer data, most notably (see Figs. 1-3):

a) The data by Poenitz and Tamura (12) and by Boldeman et al. (15) show negative deviations of 5-10% from a Maxwellian (T = 1.42 MeV) shape between 0.2 MeV and 1 MeV which are supported by all new theoretical model calculations, but the data by Blinov et al. (7) and by Lajtai et al. (14) agree with the Maxwellian shape. Data by Starostov et al.(16), Boytsov and Starostov (16), and by Boldeman et al. (15) below 0.2 MeV show positive deviations.

b) Positive deviations of ~ 3% relative to the Maxwellian shape seem to center around 2-3 MeV for the data by Poenitz and Tamura (12) and by Blinov et al. (17), but appear to center between 3-5 MeV and 3-7 MeV for the data by Boldeman et al. (15) and by Bench and Jasicek (8), respectively.

The high energy part (> 10-15 MeV) is of lesser importance for practical applications but of interest for a theoretical understanding of the fission process. First measurements up to 30 MeV were made by Maerten et al. (11) and show an excess of neutrons above a Maxwellian shape above ~ 15 MeV. These data have been recently confirmed by Boettger et al. (18), however, data uncertainties are still very large (80-100%) and measurements by Chalupka et al. (19) contradict these results.

Theoretical Models

The fission neutron spectrum has usually been represented as a Maxwellian or as a Watt spectrum. Adjustments to a Maxwellian shape in several energy segments were obtained by Grundl and Eisenhauer (22) based on measured average cross sections which resulted in an unphysical representation of the fission spectrum (see Figs. 1-2). Substantial progress in the theoretical description of fission neutron spectra has been made with the development of the Madland and Nix model (23). Though this model contains some rough approximations, its great utility lies in the easy application for the calculation of various features of the fission process (spectrum, n(E) for first and second chance fission) and the good agreement with experimental data which can be achieved by adjusting a single (level density) parameter. The Madland-Nix model has been improved by Maerten and Seeliger (20, 24) by removing some of the approximations (e.g. the A dependence of the temperature, fragment energy, and the inverse reaction cross section were taken into account, and the calculated spectra were weighted by the fragment and neutron yields). Another model, the Complex Cascade Evaporation Model, was developed by Maerten and Seeliger (20, 24). This model takes into account more detailed features of the fission and neutron emissions processes but consequently

depends on a larger number of parameters which are in part less well known than those involved in the Madland and Nix model. Hauser Feshbach calculations of the fission neutron spectrum (25, 26) which should be the best current theoretical approach have a similar problem. The various models are discussed in some detail in the excellent review by Maerten and Seeliger (20), in which comparisons between the spectra calculated with the models and some of the experimental data have also been made (see fig. 3). Based on this figure several observations can be made:

a) The agreement between the various newer models is very good over most of the energy range (± 2-3%) with differences becoming larger at lower (<0.2 MeV) and higher (> 10 MeV) energies. All models show a similar deviation from the Maxwellian spectrum shape.

b) The agreement between the recent experimental data and the newer models is good over a large energy range, except below 1 MeV where data discrepancies exist. It appears that none of the new models support the Maxwellian shape or the surplus of neutrons observed in this energy region in some of the experiments.

These observations must be taken with caution for several reasons:

a) Recent work by Froehner (27) established that essentially all recent measurements are very well described by a Watt spectrum shape (see Fig. 4), i.e. the kinetics of neutron evaporation from fully accelerated fragments overwhelms all other "microscopic" effects.

b) None of the experimental data are sufficiently accurate (<±2%) to differentiate among any of the available models, specifically at higher energies where uncertainties of the experimental data become rather large.

c) The models neglect the scission neutron component which has been estimated to contribute ~10% to the neutron emission spectra but is not well known.

Integral Data and Evaluations

Experimental data for cross sections averaged over the ^{252}Cf fission neutron spectrum are available for many dosimetry reactions and have typical uncertainties of > 4% (28, 29). An exception is the average ^{235}U(n,f) cross section which is known with ~1-2% uncertainty. Such data have been used for the testing of the various representations of the ^{252}Cf spectrum (30-33). Comparisons of the NBS (22), Watt, Maxwellian and Madland-Nix spectra using ENDF/B-V cross sections show that the Madland-Nix spectrum results in the best agreement with the experimental integral data (31, 32).

Another comparison using ENDF/B-V and JENDL-2 cross sections between a Maxwellian (E = 2.13 MeV), the Madland-Nix spectrum, the Generalized

Madland-Nix spectrum, the Complex Evaporation Model and the Hauser-Feshbach calculations show all to give similar results with the exception of the Maxwellian spectrum (33).

The new and improved differential measurements of the ^{252}Cf spectrum have resulted in three new evaluations. Mannhart (34) evaluated some of the more recent experimental data by generalized least squares fitting and applied weighted spline fitting for interpolation. The evaluation is in very good agreement with the theoretical spectrum obtained by Maerten and Seeliger (20,24). The evaluation resulted in spectrum uncertainties of ≤ 3 % between 150 keV and 11 MeV and ≤ 1.5 % between 1 and 5 MeV. Outside the quoted ranges the uncertainties increase up to 10 % at the lowest and 30 % at the highest energies. Bajkov and Yurevich (35) evaluated the difference from a Maxwellian using simultaneously differential and integral data. The evaluation resulted in spectrum uncertainties of 1-2% between ~500 keV and 9 MeV. Finally, Froehner (27) fitted the more recent experimental data with a Watt spectrum and obtained $T_W = 1.175 \pm 0.005$ MeV and $E_W = 0.359 \pm 0.009$ MeV. A fit to the evaluated data by Mannhart resulted in nearly identical parameters. Deviations of the Watt spectrum from the evaluated data are about the size of the uncertainties.

Conclusions and Recommendations

Recent experimental data have substantially reduced the uncertainty with which the ^{252}Cf spectrum is known. New theoretical models provide improved physical representations for the spectrum shape. The presently available spectra calculated with the Madland-Nix model or the Complex Evaporation Model have been transmitted to the Nuclear Data Section of the International Atomic Energy Agency in Vienna based on a recommendation of the Advisory Group Meeting on Nuclear Standard Reference Data (4) and are available on request.

The NEANDC and INDC Standards Subcommittees recommend the use of the Mannhart's evaluation of the ^{252}Cf fission neutron spectrum as standard reference data. The data are available on magnetic tape from the neutron data centers. A documentation accompanying the tape is available from the IAEA Nuclear Data Section as IAEA-NDS-98, which includes listings of the data and a reprint of W. Mannhart's description of his evaluation presented at the IAEA advisory group meeting on properties of neutron sources, Leningrad, USSR, 9-13 June 1986.

The evaluation covers the energy range from 1 keV to 20 MeV. The data are given in an ENDF similar format, in two files.

The first file gives in MF=5 the data pointwise as function of energy, with the energy points chosen dense enough for logarithmic interpolation below 750 keV and linear interpolation above.

The second file gives in MF=35 the absolute covariance matrix of the neutron spectrum in 70 energy bins (ENDF-VI format). To allow the derivation of the relative covariance matrix, there is an "auxiliary file" which gives the spectrum averages of the 70 energy bins. With these data the correlation matrix shown in the "documentation file" can be derived.

Recently, Mannhart's spectrum and its covariance matrix have been included in MAT 9861 of the ENDF/B-VI Decay Data File (Tape 200). The original data have been modified to meet the the ENDF interpolation rules (36).

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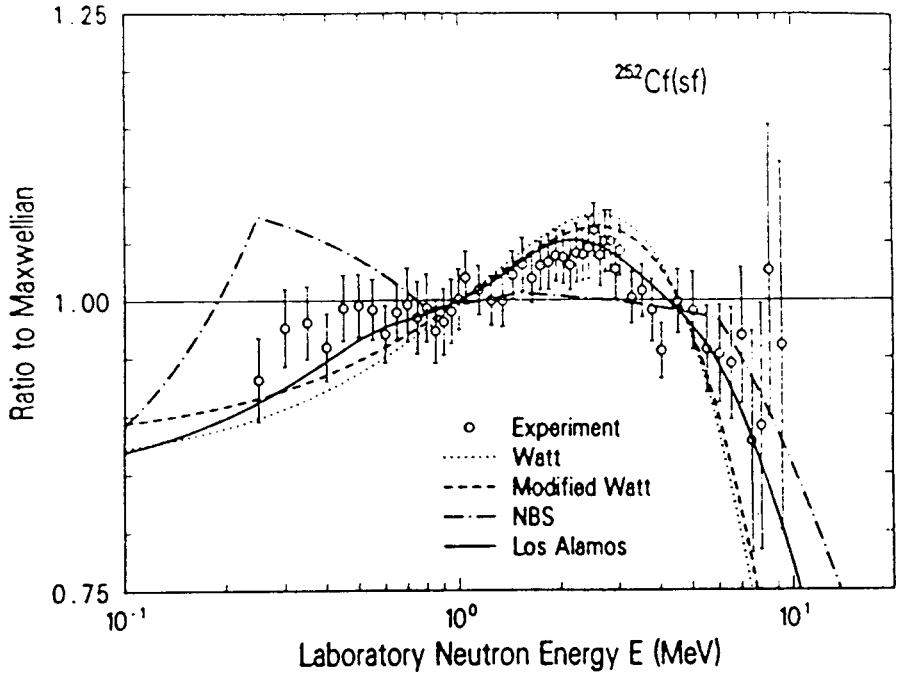

Figure 1 Differential Spectrum Comparisons for Adjustments to the Poenitz and Tamura Experiment (Ref. 12). From Ref. 31.

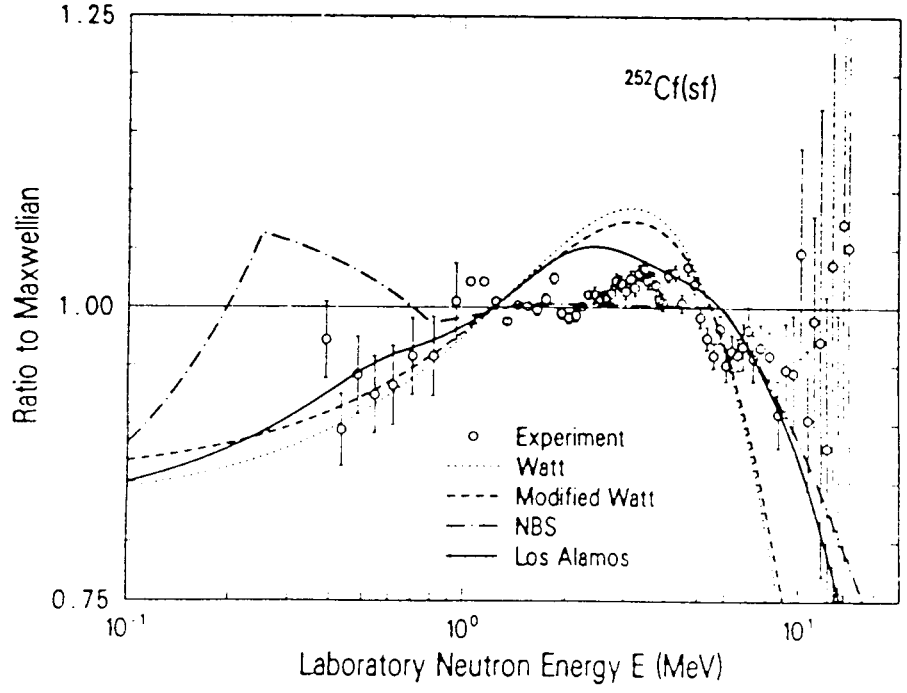

Figure 2 Differential Spectrum Comparisons for Adjustments to the Boldeman et al. Experiment (Ref. 15). From Ref. 31.

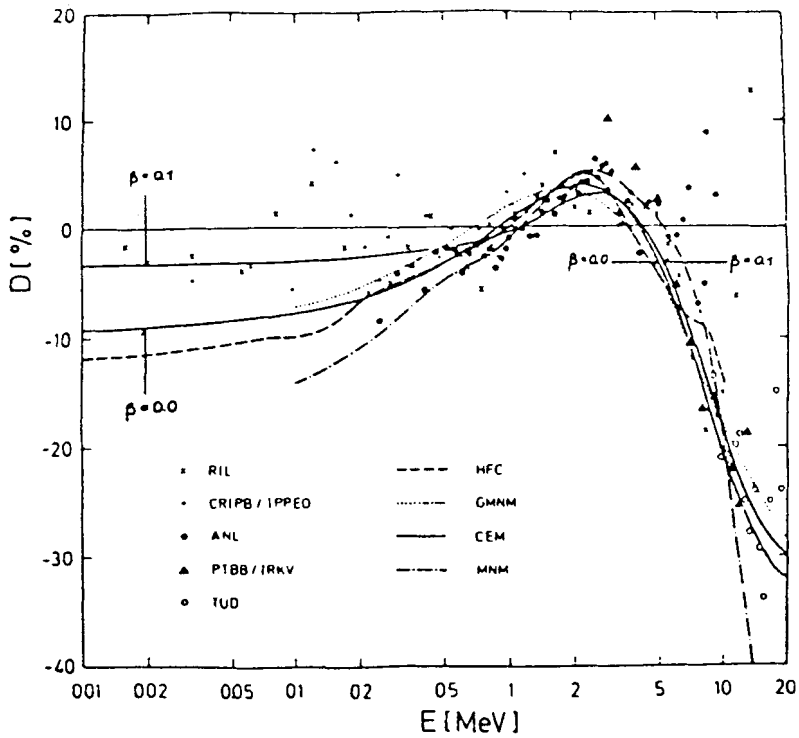

Figure 3 Comparison of Theoretical Calculations (HFC = Hanser Feshbach, GMNM = Generalized Madland-Nix, CEM = Complex Evaporation, MNM = Madland-Nix) with Experimental Data. From Ref. 20.

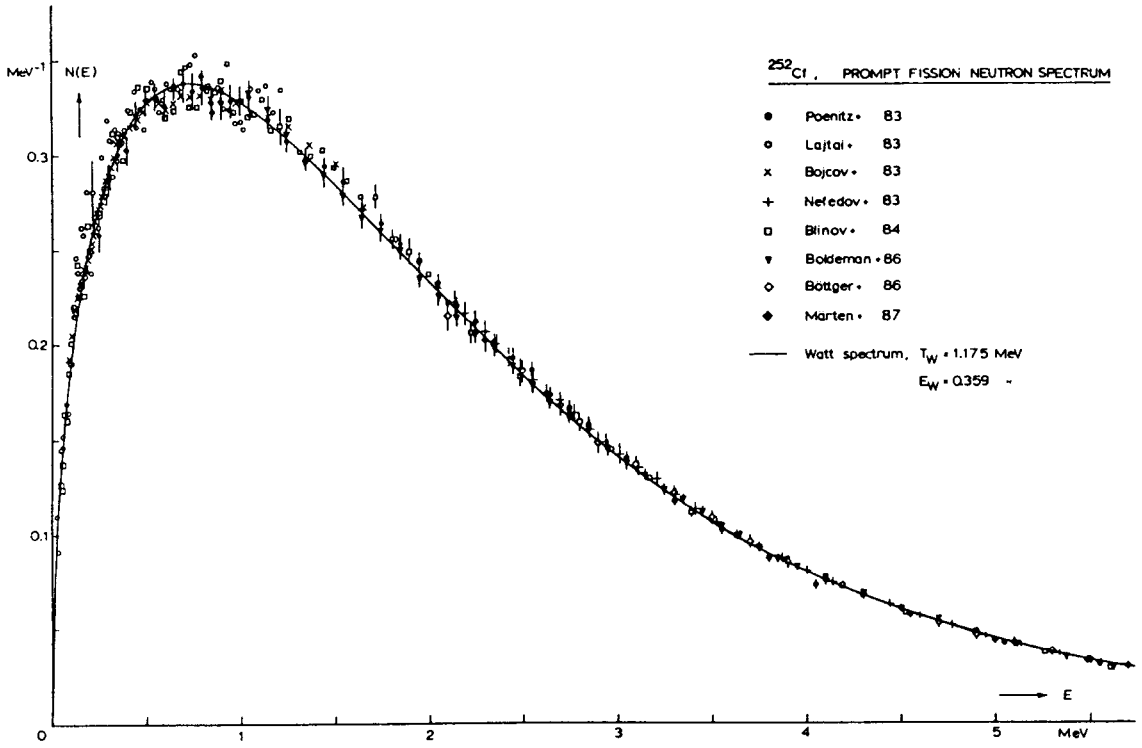

Figure 4. Fit of recent experimental data with a Watt spectrum (from Ref. 27).

NU-BAR OF ^{252}Cf

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The average number of neutrons emitted in the spontaneous fission of ^{252}Cf ($\bar{\nu}$ for total neutron emission, $\bar{\nu}_p$ for prompt neutron emission) has been the reference standard for the majority of $\bar{\nu}$ measurements and has also been applied as a standard in a number of neutron emission rate experiments. A comprehensive listing of all determinations of this standard is given in Table I. Preliminary data from White and Axton (6), Axton et al. (7) and Aleksanderov et al. (9) have been superseded by subsequent values. The only new measurement in the table is that of Edwards et al. (15). A recent evaluation of all $\bar{\nu}$ measurements by Axton (16) which included a comprehensive assessment of all common sources of error, arrived at a recommended value of 3.7661 ± 0.0054 for $\bar{\nu}$ for ^{252}Cf.

Recently there has been renewed interest in the neutron emission probability distribution P_ν for ^{252}Cf because of its application in safeguards measurements. Table II lists a set of recommended values for P_ν. These values have been obtained by incorporating the recent data from Boldeman and Hines (20) into the evaluation of Holden and Zucher (2).

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Table I. Nu-bar Values for Spontaneous Fission of ^{252}Cf

Experiment	Value
Liquid Scintillator	
*Asplund-Nilsson et al (1963) (1)	3.792±0.040
*Hopkins and Diven (1963) (2)	3.777±0.031
*Boldeman (1974) (3)	3.755±0.016
Zhang and LIU (1980) (4)	3.754±0.018
Spencer et al (1982) (5)	3.782 ± 0.007
Manganese Bath	
White and Axton (1968) (6)	superseded
Axton et al (1969) (7)	superseded
**De Volpi and Porges (1970) (8)	3.747±0.019
Aleksandrov et al (1975) (9)	superseded
Bozorgmanesh (1977) (10)	3.744±0.023
Aleksandrov et al (1980) (11)	3.758±0.015
Smith and Reeder (1984) (12)	3.767±0.011
Axton and Bardell (1984) (13)	3.7509±0.0107
Boron Pile	
***Colvin and Sowerby (1965) (14)	3.739±0.021
Edwards et al (1982) (15)	3.761±0.029
Evaluation	
Axton (1984) (16)	3.7661±0.0054
ENDF/B-VI	3.7676±0.0049

* Revised by Boldeman (1977) (17)
** Revised by Smith (1977) (18)
*** Revised by Ullo (1977) (19)

Table II Neutron Emission Parameters for Spontaneous Fission of ^{252}Cf

ν_p	3.757
P_0	0.00219
P_1	0.02608
P_2	0.12600
P_3	0.27374
P_4	0.30399
P_5	0.18493
P_6	0.06639
P_7	0.01473
P_8	0.00188
P_9	0.00007
$\langle\nu^2\rangle_{av}$	15.719
$\sigma^2(\nu)$	1.6034
R	0.8474

$\langle\nu^2\rangle_{av} = \sum \nu^2 P_\nu$ $\qquad$ $\sigma^2(\nu) = \sum P_\nu(\nu - \bar{\nu})^2$ $\qquad$ $R = (\langle\nu^2\rangle_{av} - \bar{\nu}) / (\bar{\nu})^2$

NEUTRON FLUX COMPARISON

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Introduction

International last neutron fluence rate intercomparisons between standardizing laboratories of many countries have been organized under the auspices of the "Comité Consultatif pour les Etalons de Mesure des Rayonnements Ionisants / Comité International des Poids et Mesures". (CCEMRI/CIPM).
The list of participants is given in Annex I.
The actual status of comparisons is presented:

1) Transfer method using ^{115}In (n, γ) ^{116m}In reaction at 144 keV and 565 keV: Coordinator : T.B. RYVES (NPL).

The six laboratories which took part in this intercomparison were CBNM, ETL, IAE, NPL, NRC and PTB. The 4πb proportional counter, medium foils and ^{60}Co check source were returned to NPL after each measurement. The fluence rates were intercompared by measuring the activation of the medium foils by means of quasi-monoenergetic neutrons of approximately 144 keV and 565 keV. In all cases the ^{7}Li (p, n) reaction was employed as the source of neutrons.
The results were expressed as the ratio:

$$R = \frac{\text{Saturated specific } \beta \text{ activity of } ^{116\,m}\text{In} \times 10^{-4}}{\text{neutron fluence rate}} \quad \text{cm}^2\,\text{g}^{-1}$$

The results of this fluence intercomparison, R, are given in Table 1. In the first column at each energy are the "original" results as first submitted by each laboratory. The "revised" results of NPL and PTB at 144 keV which were received at a later date are also given and finally the "consistent" results, which have been obtained by treating the target scatter and $^{115\,m}In$ corrections in a unified manner for all the laboratories, thus eliminating some of the uncorrelated uncertainties. This should reduce the experimental spread of the intercomparison.

The total 1σ uncertainties (%) are also given in table 1.
At 144 keV the results showed a spread of 4 % (ignoring NRC) and at 565 keV 9 %.

The intercomparison demonstrated that the target scattering corrections were not consistent, especially at 144 keV and that the ^{115}In (n,γ) transfer device was very sensitive to the low energy neutron components from room and target-scatter due to the increase in the In capture cross-section.

2) Transfer method using ^{115}In (n, n') ^{115m}In reaction at 2,5 MeV, 5 MeV and 14,8 MeV: Coordinator : H. LISKIEN (CBNM)

Previous results of the comparison at 2,5 MeV and 5 MeV for seven laboratories are given in figure 1.

We illustrate the general consistency for this results at 2,5 MeV and 5 MeV and for the results concerning ^{93}Nb (n, 2n) ^{92m}Nb at 14,8 MeV as shown on the same figure 1.

Concerning measurements at 14,8 MeV based on ^{115}In (n, n') ^{115m}In, the normalisation of the results, submitted by nine participants, to a common energy and the corrections from room scattered and target scattered neutrons are still in progress at PTB.

3) Fast neutron rate comparison using fission chamber instruments : Coordinator : D.B. GAYTHER (AERE)

Since the last meeting of "CCEMRI" (section III), measurements have been made with the Harwell fission chambers at CBNM, ETL and NPL, bringing the number of laboratories to six who have participated in this intercomparison. Table II summarizes the present state of the programme.

A preliminary assessment of the results is shown in figures 2, 3 and 4.

Since the various laboratories measured the chamber sensitives at slightly different neutron energies, a correction has been applied to normalize all the results to the standard energies (0.144, 0.565, 2.5, 5.0, 14.8 MeV). The correction was based on the energy-dependence of the fission cross-sections given in the ENDF/B-VI evaluation.

The calculated sensitivity at each measurement energy was obtained from the accurately determined mass of the fissile content of the chamber (known only to the coordinator) and the ENDF/B-VI evaluated cross-sections corrections were applied for neutron scattering in the multiplate arrangement and for loss of fission fragments in the deposit.

It can be seen from the figures 2, 3, 4 that there are no large discrepancies in the measurements and that the calculated sensitivities agree with the observations.

Conclusion

As we see, the different intercomparisons started a few years ago are now quite complete. There still remain normalization and correction problems especially due to scattered neutrons. The section III of the "CCEMRI" and the working groups associated continue to consider transportable transfer instruments such as:

- moderating spheres at all energies
- ^{3}He spectrometer up to 250 keV
- proportional counter proton recoil spectrometer up to 500 keV
- NE 213 scintillator proton recoil spectrometer above 500 keV

and to search for suitable activations for other neutron energies.

ANNEX I - Participants

AERE	:	Atomic Energy Research Establishment, Harwell, UK
BARC	:	Bhabha Atomic Research Centre, Bombay, India
BIPM	:	Bureau International des Poids et Mesures, Sevres, France
CBNM	:	Central Bureau for Nuclear Measurements, Geel, Belgium
ETL	:	Electrotechnical Laboratory, Ibaraki, Japan
IAE	:	Institute of Atomic Energy, Beijing, PR China
IMM	:	Institut de Metrologie Mendeleev, Leningrad, URSS
IRK	:	Vienna, Austria
NBS	:	National Bureau of Standards, Washington, USA
NPL	:	National Physical Laboratory, Teddington, UK
NRC	:	National Research Council of Canada, Ottawa, Canada
PTE	:	Physikalisch-Technische Bundesanstalt, Braunschweig, FR Germany

Neutron energy / Laboratory	144 KeV		565 KeV	
	Original (revised)	Consistent	Original (revised)	Consistent
CBNM	-	-	5,33	5,35 (1,9 %)
ETL	7,46	7,63 (2,9 %)	5,15	5,19 (2,3 %)
IAE	7,36	7,45 (3,2 %)	5,23	5,24 (2,9 %)
NPL	7,00 (7,31)	7,38 (2,4 %)	4,98	4,94 (2,2 %)
NRC	9,86	9,56 (5,6 %)	5,31	5,26 (3,6 %)
PTB	7,79 (7,41)	7,41 (3,1 %)	4,91 (4,93)	4,94 (2,8 %)

Table 1

Results of the intercomparison for $^{115}In\ (n,\gamma)\ ^{116m}In$

R : ratio of the saturated specific β activity of ^{116m}In to the neutron fluence x 10^{-4} $cm^2.g^{-1}$

Table II

Summary of fluence intercomparison

LABORATORY	DATES	ENERGIES (MeV)	ACCELERATOR	FLUENCE MONITOR	COMMENTS
NBS+	Jan.-July 1983	0.53	Linac, Pulsed Van de Graaff	Black detector	Only ^{235}U chamber used. Excellent agreement between Linac and Van de Graaff measurements. Final report written.
BIPM+	Feb.-May 1984	14.65	Unpulsed D-T generator	Associated particle	Both chambers used. Preliminary report written.
PTB+	Oct. 1984 - March 1985	2.5,5.0, 14.65	Pulsed Van de Graaff	Proton recoil telescope	Both chambers used. Final report written.
CBNM+	April 1985 - February 1986	All	Linac, Pulsed Van de Graaff	Proton recoil telescope and proportional counter	Van de Graaff measurements completed with both chambers at 0.565, 2.5, 5.0, 14.8 MeV and final report written. Linac not yet used.
ETL+	Spring 1986	0.144, 0.565, 5.0,14.6	Cockcroft-Walton Pelletron		Both chambers used. Final report written.
NPL+	Spring 1987	0.565, 14.7	Pulsed Van de Graaff, Sames generator	Long Counter, Associated particle and foil activation	14.7 MeV measurement completed with both chambers and report written.
HAR+		All	Linac, Pulsed Van de Graaff?		Last measurement.

+NBS: National Bureau of Standards (USA), BIPM: Bureau International des Poids et Mesures (France), PTB: Physikalische Technische Bundesanstalt (FRG), CBNM: Central Bureau for Nuclear Measurements (EEC, Belgium), ETL: Electrotechnical Laboratory (Japan), NPL: National Physical Laboratory (UK), HAR: Harwell Laboratory (UK).

Figure 1

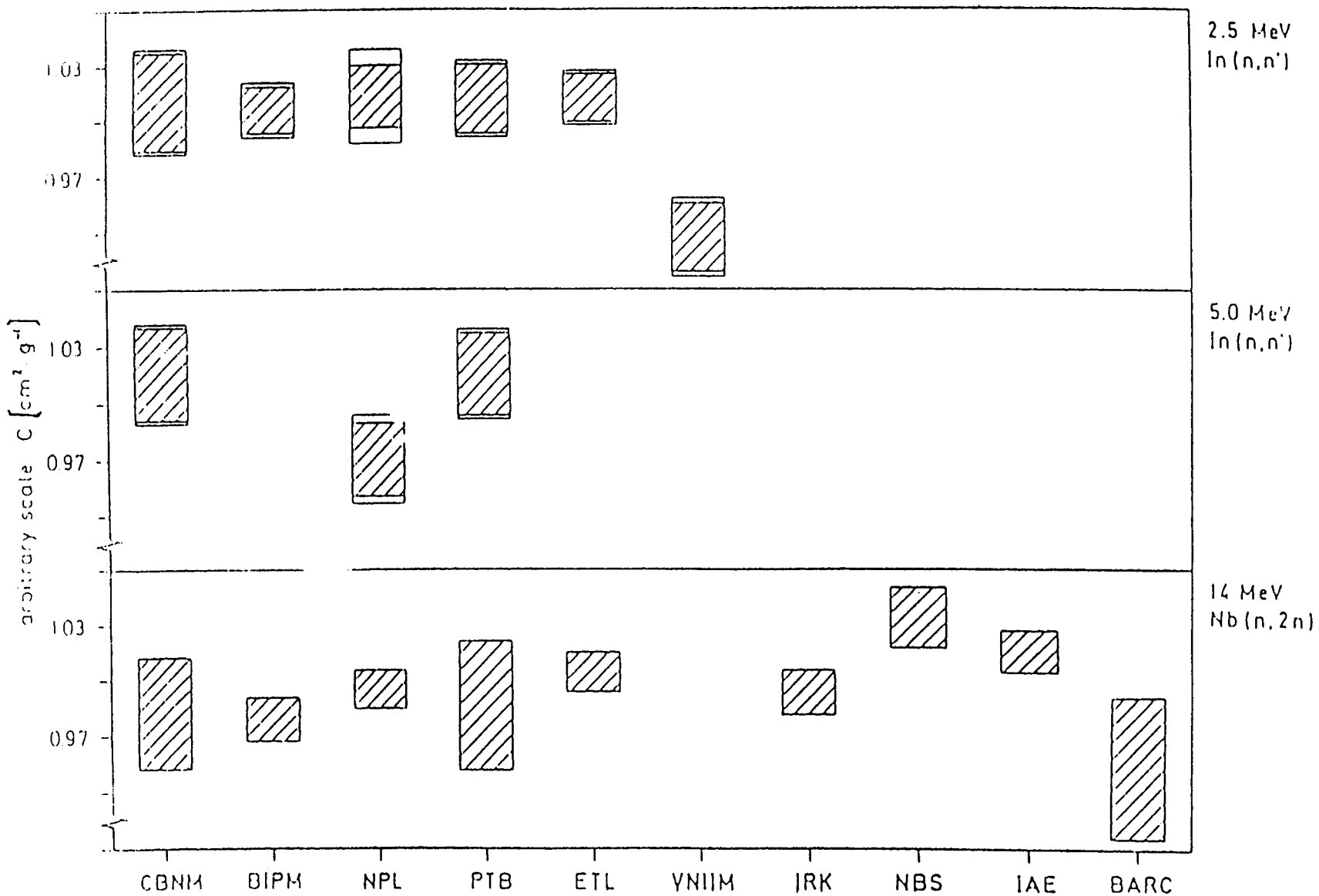
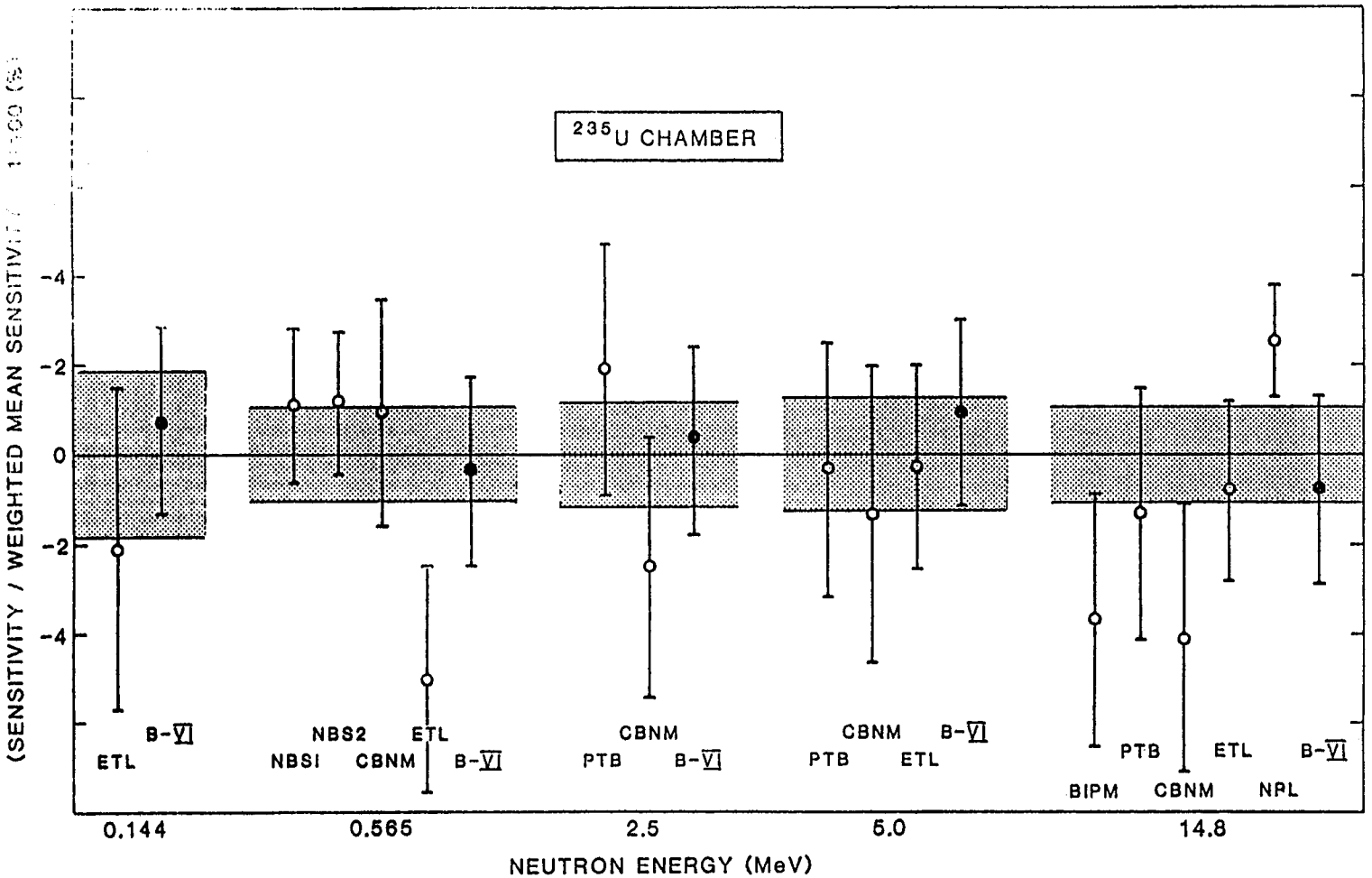
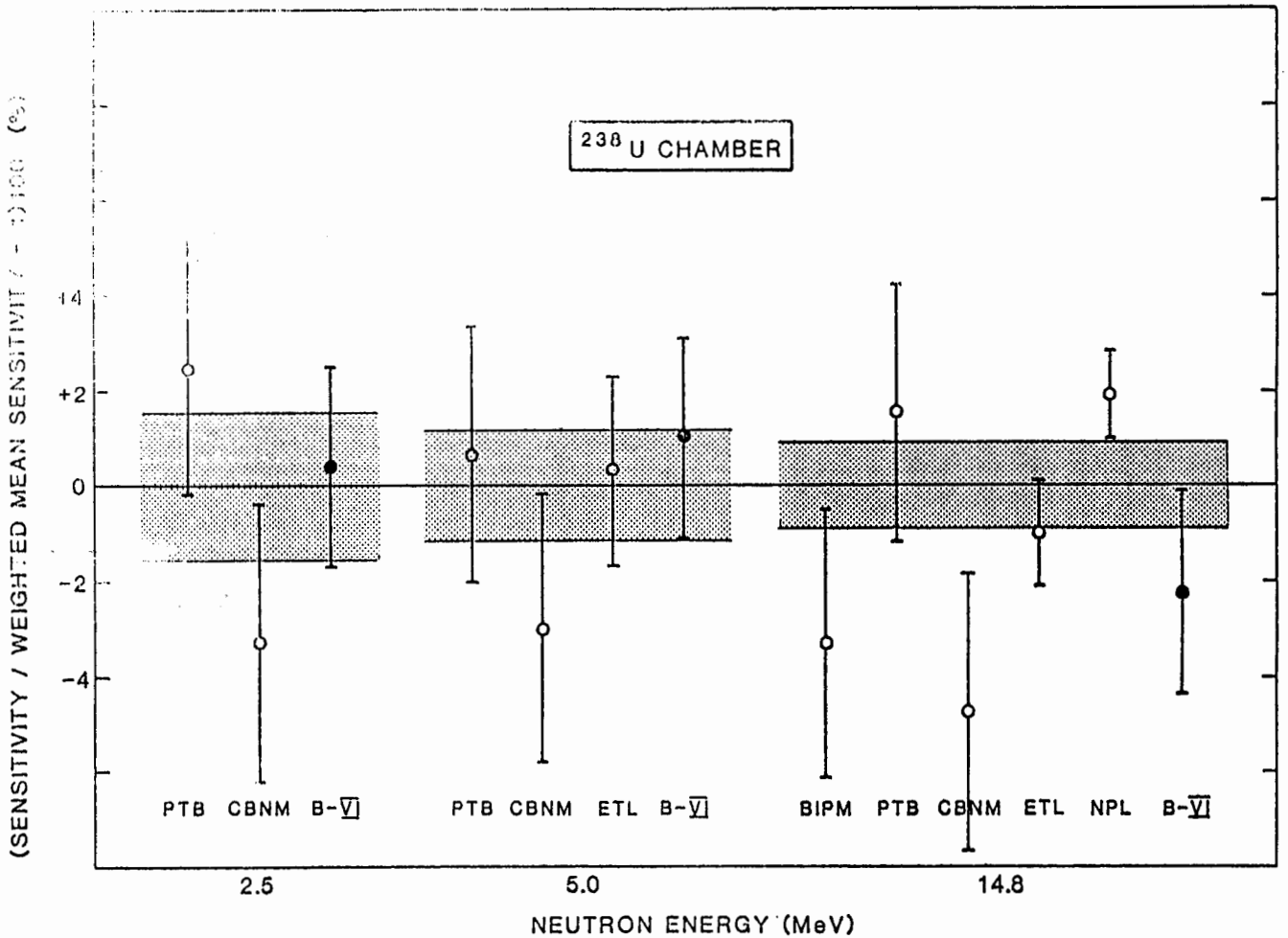

2.5 MeV
In (n,n')
5.0 MeV
In (n,n')
14 MeV
Nb (n,2n)
1.03
0.97
1.03
0.97
1.03
0.97
arbitrary scale C [cm² g⁻¹]
CBNM
BIPM
NPL
PTB
ETL
VNIIM
IRK
NBS
IAE
BARC

Figure 2



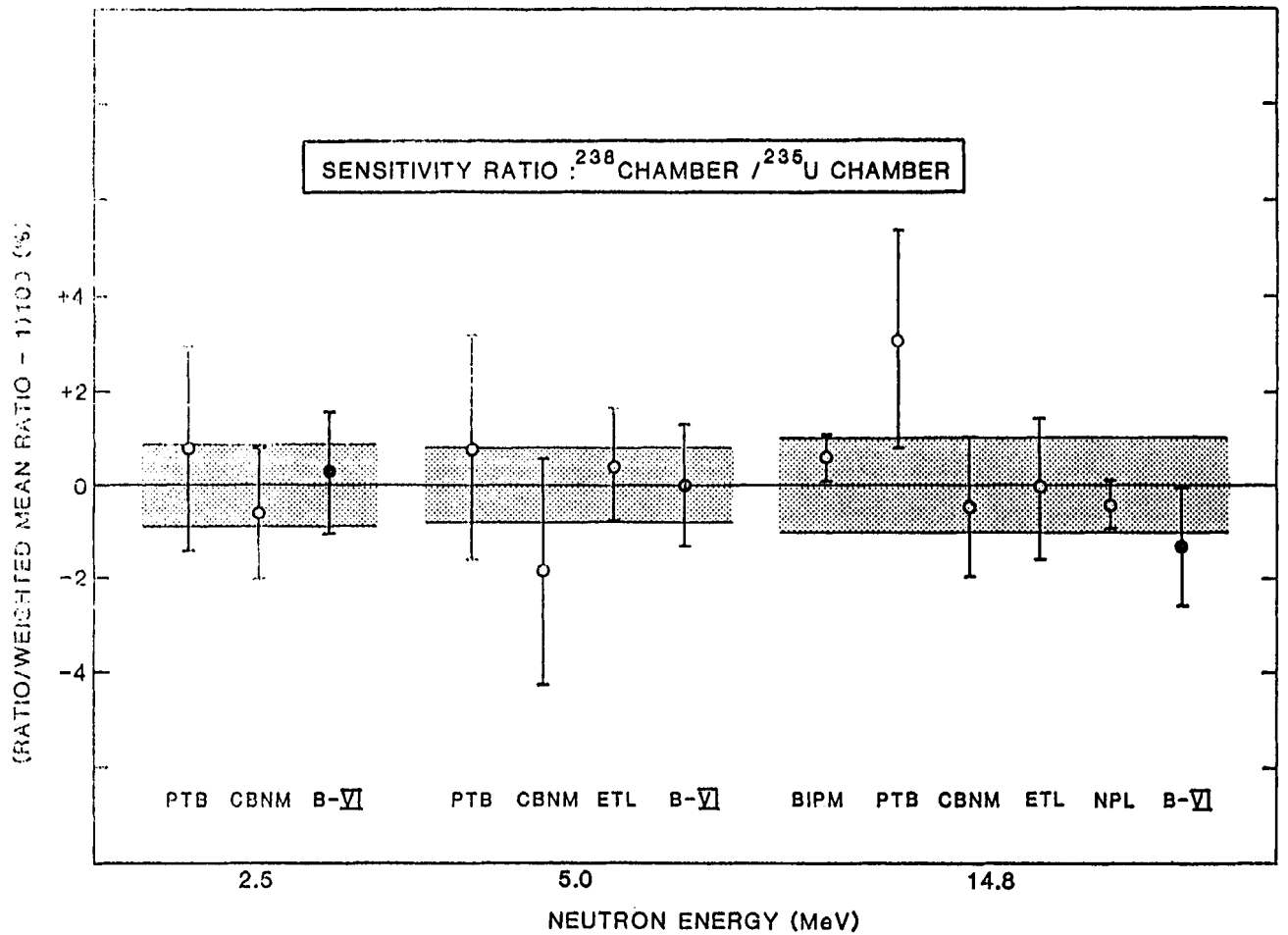
Preliminary comparison of measured fission chamber neutron sensitivities (open circles). The solid circles are the calculated sensitivities based on the ENDF/B-VI fission cross-section and the known mass of ^{235}U in the chamber. Results are expressed as the percentage difference from the weighted mean sensitivity.

Figure 3



Preliminary comparison of measured fission chamber neutron sensitivities (open circles). The solid circles are the calculated sensitivities based on the ENDF/B-VI fission cross-section and the known mass of ^{238}U in the chamber. Results are expressed as the percentage difference from the weighted mean sensitivity.

Figure 4



Preliminary comparison of measured ratios of the neutron sensitivities of the two fission chambers (open circles). The solid circles are calculated sensitivity ratios derived from the ENDF/B-VI fission cross-sections and the known masses of ^{235}U and ^{238}U in the chambers. Results are expressed as the percentage difference from the weighted mean sensitivity ratio.

X-RAY AND GAMMA-RAY STANDARDS

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June 1988

Updated by H D Lemmel, IAEA, Vienna, August 1992

The efficiency calibration of gamma-ray detectors requires a precise knowledge of the gamma-ray energies and emission probabilities of the calibrant radionuclides. The need for an universally accepted base of radionuclide decay data to serve as a standard for the efficiency calibration of gamma-ray detectors has become increasingly apparent in the course of nuclear measurement intercomparison programmes performed over the past few years. A variety of reference data sets has been developed for this purpose in many gamma spectroscopy laboratories. The formulation and use of a single internationally produced and accepted file of carefully evaluated decay data would eliminate inconsistencies and improve the accuracy of detector efficiency calibrations.

In 1986, the IAEA initiated a Coordinated Research Programme (CRP) aimed specifically at the production of a single internationally accepted set of x-ray and gamma-ray detector calibration data of improved quality to meet the needs of radioactivity measurements in fields such as safeguards, dosimetry and fuel management. In particular, this programme examines the current status and adequacy of radionuclide decay data used for detector efficiency calibration, identifies additional nuclides which could be appropriate as calibration standards, and initiates appropriate actions, (i.e. required measurements and/or evaluations) to produce the required file of calibration data.

The conclusions of this CRP have been published in IAEA-TECDOC-619 (Sept 1991). The recommended values from this report are listed on the following pages. These data are also available on a PC diskette by Hartmut Lemmel which can be obtained from the IAEA Nuclear Data Section, cost-free upon request.

References

Work referred to in the lists of Recommended Reference Data was performed in the context of the IAEA Coordinated Research Programme on X- and Gamma-ray Standards for Detector Efficiency Calibration (1986 - 1989)

1. M J Woods, A S Munster, National Physical Laboratory (NPL), Teddington, Middlesex, UK and K Debertin, Physikalisch-Technische Bundesanstalt (PTB),Braunschweig, Germany

References (cont)

2. K Debertin, Physikalisch-Technische Bundesanstalt (PTB), Braunschweig, Germany and M J Woods, A S Munster, National Physical Laboratory (NPL), Teddington, Middelsex, UK
3. W Bambynek, CEC-JRC, Central Bureau for Nuclear Measurements (CBNM), Geel, Belgium
4. F J Schima, National Institute of Standards and Technology (NIST), Gaithersburg, Maryland, USA
5. Y Yoshizawa, Faculty of Sciences, Hiroshima University, Hiroshima-Shi, Japan
6. A L Nichols, AEA Technology, Winfrith Technology Centre, Dorchester, Dorset, UK
7. T Barta, R Jedlovszky, National Office of Measures (OMH), Budapest, Hungary
8. N Coursol, Laboratoire de Métrologie des Rayonnements Ionisants (LMRI), Gif-sur-Yvette, France
9. R G Helmer, Idaho National Engineering Laboratory (INEL), Idaho Falls, Idaho, USA
10. F Lagoutine, Laboratoire de Métrologie des Rayonnements Ionisants (LMRI), Gif-sur-Yvette, France

X-RAY AND GAMMA-RAY STANDARDS - Recommended Reference Data

Table 1: Half-Lives of Radionuclides Used for Detector Calibration

Nuclide	Decay Mode	Half-life (days) Value	Uncertainty	Exponent	Reference
11-Na-022	EC	950.8	± 0.9		(1)
11-Na-024	β-	0.62356	± 0.00017		(1)
21-Sc-046	β-	83.79	± 0.04		(1)
24-Cr-051	EC	27.706	± 0.007		(1)
25-Mn-054	EC	312.3	± 0.4		(1)
26-Fe-055	EC	999	± 8		(1)
27-Co-056	EC	77.31	± 0.19		(1)
27-Co-057	EC	271.79	± 0.09		(1)
27-Co-058	EC	70.86	± 0.07		(1)
27-Co-060	β-	1925.5	± 0.5		(1)
30-Zn-065	EC	244.26	± 0.26		(1)
34-Se-075	EC	119.64	± 0.24		(1)
38-Sr-085	EC	64.849	± 0.004		(1)
39-Y-088	EC	106.630	± 0.025		(1)
41-Nb-093m	IT	5890	± 50		(2)
41-Nb--094	β-	7.3	± 0.9	E+06	(2)
41-Nb-095	β-	34.975	± 0.007		(2)
48-Cd-109	EC	462.6	± 0.7		(2)
49-In-111	EC	2.8047	± 0.0005		(2)
50-Sn-113	EC	115.09	± 0.04		(2)
51-Sb-125	β-	1007.7	± 0.6		(2)
53-I-125	EC	59.43	± 0.06		(2)
55-Cs-134	β-	754.28	± 0.22		(2)
55-Cs-137	β-	1.102	± 0.006	E+04	(2)
56-Ba-133	EC	3862	± 15		(2)
58-Ce-139	EC	137.640	± 0.023		(2)
63-Eu-152	EC	4933	± 11		(2)
63-Eu-154	β-	3136.8	± 2.9		(2)
63-Eu-155	β-	1770	± 50		(2)
79-Au-198	β-	2.6943	± 0.0008		(2)
80-Hg-203	β-	46.595	± 0.013		(2)
83-Bi-207	EC	1.16	± 0.07	E+04	(2)
90-Th-228	α	698.2	± 0.6		(1)
93-Np-239	β-	2.350	± 0.004		(2)
95-Am-241	α	1.5785	± 0.0024	E+05	(2)
95-Am-243	α	2.690	± 0.008	E+06	(1)

X-RAY AND GAMMA-RAY STANDARDS - Recommended Reference Data

Table 2: X-Ray Standards, Energies and Emission Probabilities

The data uncertainties are standard deviations. For the emission probabilities uncertainties are noted as last digit uncertainties, i.e. 12.3(4) means 12.3 ± 0.4 and 12.3(14) means 12.3 ± 1.4

Nuclide	Trans	Energy (keV)	Probability
24-Cr-051	VKα	4.95	0.201(3)
24-Cr-051	VKβ	5.43	0.027(1)
24-Cr-051	VKx	4.95-5.43	0.228(3)
25-Mn-054	CrKα	5.41	0.226(7)
25-Mn-054	CrKβ	5.95	0.030(1)
25-Mn-054	CrKx	5.41-5.95	0.256(8)
26-Fe-055	MnKα	5.89	0.249(9)
26-Fe-055	MnKβ	6.49	0.034(1)
26-Fe-055	MnKx	5.89-6.49	0.283(10)
27-Co-057	FeKα	6.40	0.510(7)
27-Co-057	FeKβ	7.06	0.069(1)
27-Co-057	FeKx	6.40-7.06	0.579(8)
27-Co-058	FeKα	6.40	0.235(3)
27-Co-058	FeKβ	7.06	0.032(1)
27-Co-058	FeKx	6.40-7.06	0.267(3)
30-Zn-065	CuKα	8.03-8.05	0.341(6)
30-Zn-065	CuKβ	8.91	0.046(1)
30-Zn-065	CuKx	8.03-8.91	0.387(6)
34-Se-075	AsKα	10.51-10.54	0.493(11)
34-Se-075	AsKβ	11.72-11.95	0.075(2)
34-Se-075	AsKx	10.51-11.95	0.568(13)
38-Sr-085	RbKα	13.34-13.40	0.500(3)
38-Sr-085	RbKβ	14.96-15.29	0.087(2)
38-Sr-085	RbKx	13.34-15.29	0.587(4)

Table 2 (cont)

Nuclide	Trans	Energy (keV)	Probability
39-Y-088	SrKα	14.10-14.17	0.522(6)
39-Y-088	SrKβ	15.83-16.19	0.094(2)
39-Y-088	SrKx	14.10-16.19	0.616(7)
41-Nb-093m	NbKα	16.52-16.62	0.0925(30)
41-Nb-093m	NbKβ	18.62-19.07	0.0179(7)
41-Nb-093m	NbKx	16.52-19.07	0.1104(35)
48-Cd-109	AgKα	21.99-22.16	0.821(9)
48-Cd-109	AgKβ	24.93-25.60	0.173(3)
48-Cd-109	AgKx	21.99-25.60	0.994(10)
49-In-111	CdKα	22.98-23.17	0.684(5)
49-In-111	CdKβ	26.09-26.80	0.146(3)
49-In-111	CdKx	22.98-26.80	0.830(5)
50-Sn-113	InKα	24.00-24.21	0.796(6)
50-Sn-113	InKβ	27.27-28.02	0.172(3)
50-Sn-113	InKx	24.00-28.02	0.968(6)
53-I-125	TeKα	27.20-27.47	1.135(21)
53-I-125	TeKβ	30.98-31.88	0.255(6)
53-I-125	TeKx	27.20-31.88	1.390(25)
55-Cs-137	BaKα	31.82-32.19	0.0566(16)
55-Cs-137	BaKβ	36.36-37.45	0.0134(5)
55-Cs-137	BaKx	31.82-37.45	0.0700(20)
56-Ba-133	CsKα	30.63-30.97	0.980(14)
56-Ba-133	CsKβ	34.97-36.01	0.230(5)
56-Ba-133	CsKx	30.63-36.01	1.210(16)
58-Ce-139	LaKα	33.03-33.44	0.643(18)
58-Ce-139	LaKβ	37.78-38.93	0.154(5)
58-Ce-139	LaKx	33.03-38.93	0.797(22)

Table 2 (cont)

Nuclide	Trans	Energy (keV)	Probability
63-Eu-152	SmKα	39.52-40.12	0.591(12)
63-Eu-152	GdKα	42.31-43.00	0.00648(22)
63-Eu-152	SmKβ	45.38-46.82	0.149(3)
63-Eu-152	GdKβ	48.65-50.21	0.00176(18)
63-Eu-152	SmKx	39.52-46.82	0.740(12)
63-Eu-152	GdKx	42.31-50.21	0.00824(28)
63-Eu-154	GdKα	42.31-43.00	0.205(6)
63-Eu-154	GdKβ	48.65-50.21	0.051(2)
63-Eu-154	GdKx	42.31-50.21	0.256(6)
79-Au-198	HgKα	68.89-70.82	0.0219(8)
79-Au-198	HgKβ	80.12-82.78	0.0061(3)
79-Au-198	HgKx	68.89-82.78	0.0280(10)
80-Hg-203	TlLx	8.95-14.40	0.060(12)
80-Hg-203	TlKα2	70.83	0.038(2)
80-Hg-203	TlKα1	72.87	0.064(2)
80-Hg-203	TlKβ1	82.43	0.022(1)
80-Hg-203	TlKβ2	85.19	0.0063(3)
80-Hg-203	TlKx	70.83-85.19	0.130(4)
83-Bi-207	PbLx	9.19-14.91	0.325(13)
83-Bi-207	PbKα2	72.80	0.226(12)
83-Bi-207	PbKα1	74.97	0.382(20)
83-Bi-207	PbKβ1	84.79	0.130(10)
83-Bi-207	PbKβ2	87.63	0.039(3)
83-Bi-207	PbKx	72.80-87.63	0.777(26)
95-Am-241	NpLl	11.871	0.0085(3)
95-Am-241	NpLα	13.927	0.132(4)
95-Am-241	NpLβη	17.611	0.194(6)
95-Am-241	NpLγ	20.997	0.049(2)

X-RAY AND GAMMA-RAY STANDARDS - Recommended Reference Data

Table 3: Gamma Ray Standards, Energies and Emission Probabilities

The data uncertainties are standard deviations. For the emission probabilities uncertainties are noted as last digit uncertainties, i.e. 12.3(4) means 12.3 ± 0.4 and 12.3(14) means 12.3 ± 1.4

Nuclide	Energy (keV)	Probability	Reference
11-Na-022	1274.542(7)	0.99935(15)	(4)
11-Na-024	1368.633(6)	0.999936(15)	(4)
11-Na-024	2754.030(14)	0.00855(5)	
21-Sc-046	889.277(3)	0.999844(16)	(5)
21-Sc-046	1120.545(4)	0.999874(11)	
24-Cr-051	320.0842(9)	0.0986(5)	(6)
25-Mn-054	834.843(6)	0.999758(24)	(5)
27-Co-056	846.764(6)	0.99933(7)	(5)
27-Co-056	1037.844(4)	0.1413(5)	
27-Co-056	1175.099(8)	0.02239(11)	
27-Co-056	1238.287(6)	0.6607(19)	
27-Co-056	1360.206(6)	0.04256(15)	
27-Co-056	1771.350(15)	0.1549(5)	
27-Co-056	2015.179(11)	0.03029(13)	
27-Co-056	2034.759(11)	0.07771(27)	
27-Co-056	2598.460(10)	0.1696(6)	
27-Co-056	3201.954(14)	0.0313(9)	
27-Co-056	3253.417(14)	0.0762(24)	
27-Co-056	3272.998(14)	0.0178(6)	
27-Co-056	3451.154(13)	0.0093(4)	
27-Co-056	3548.27(10)	0.00178(9)	
27-Co-057	14.4127(4)	0.0916(15)	(7)
27-Co-057	122.0614(3)	0.8560(17)	
27-Co-057	136.4743(5)	0.1068(8)	
27-Co-058	810.775(9)	0.9945(1)	(7)
27-Co-060	1173.238(4)	0.99857(22)	(4)
27-Co-060	1332.502(5)	0.99983(6)	
30-Zn-065	1115.546(4)	0.5060(24)	(6)

Table 3 (cont)

Nuclide	Energy (keV)	Probability	Reference
34-Se-075	96.7344(10)	0.0341(4)	(6)
34-Se-075	121.1171(14)	0.171(1)	
34-Se-075	136.0008(6)	0.588(3)	
34-Se-075	264.6580(17)	0.590(2)	
34-Se-075	279.5431(22)	0.250(1)	
34-Se-075	400.6593(13)	0.115(1)	
38-Sr-085	514.0076(22)	0.984(4)	(5)
39-Y-088	898.042(4)	0.940(3)	(8)
39-Y-088	1836.063(13)	0.9936(3)	
41-Nb-094	702.645(6)	0.9979(5)	(9)
41-Nb-094	871.119(4)	0.9986(5)	
41-Nb-095	765.807(6)	0.9981(3)	(9)
48-Cd-109	88.0341(11)	0.0363(2)	(8)
49-In-111	171.28(3)	0.9078(10)	(5)
49-In-111	245.35(4)	0.9416(6)	
50-Sn-113	391.702(4)	0.6489(13)	(9)
51-Sb-125	176.313(1)	0.0685(7)	(8)
51-Sb-125	380.452(8)	0.01518(16)	
51-Sb-125	427.875(6)	0.297(3)	
51-Sb-125	463.365(5)	0.1048(11)	
51-Sb-125	600.600(4)	0.1773(18)	
51-Sb-125	606.718(3)	0.0500(5)	
51-Sb-125	635.954(5)	0.1121(12)	
53-I-125	35.4919(5)	0.0658(8)	(8)
55-Cs-134	475.364(3)	0.0149(2)	(5)
55-Cs-134	563.240(4)	0.0836(3)	
55-Cs-134	569.328(3)	0.1539(6)	
55-Cs-134	604.720(3)	0.9763(6)	
55-Cs-134	795.859(5)	0.854(3)	
55-Cs-134	801.948(5)	0.0869(3)	
55-Cs-134	1038.610(7)	0.00990(5)	
55-Cs-134	1167.968(5)	0.01792(7)	
55-Cs-134	1365.185(7)	0.03016(11)	
55-Cs-137	661.660(3)	0.851(2)	(8)

Table 3 (cont)

Nuclide	Energy (keV)	Probability	Reference
56-Ba-133	80.998(5)	0.3411(28)	(7)
56-Ba-133	276.398(1)	0.07147(30)	
56-Ba-133	302.853(1)	0.1830(6)	
56-Ba-133	356.017(2)	0.6194(14)	
56-Ba-133	383.851(3)	0.08905(29)	
58-Ce-139	165.857(6)	0.7987(6)	(8)
63-Eu-152	121.7824(4)	0.2837(13)	(9)
63-Eu-152	244.6989(10)	0.0753(4)	
63-Eu-152	344.2811(19)	0.2657(11)	
63-Eu-152	411.126(3)	0.02238(10)	
63-Eu-152	443.965(4)	0.03125(14)	
63-Eu-152	778.903(6)	0.1297(6)	
63-Eu-152	867.390(6)	0.04214(25)	
63-Eu-152	964.055(4)	0.1463(6)	
63-Eu-152	1085.842(4)	0.1013(5)	
63-Eu-152	1089.767(14)	0.01731(9)	
63-Eu-152	1112.087(6)	0.1354(6)	
63-Eu-152	1212.970(13)	0.01412(8)	
63-Eu-152	1299.152(9)	0.01626(11)	
63-Eu-152	1408.022(4)	0.2085(9)	
63-Eu-154	123.071(1)	0.412(5)	(5)
63-Eu-154	247.930(1)	0.0695(9)	
63-Eu-154	591.762(5)	0.0499(6)	
63-Eu-154	692.425(4)	0.0180(3)	
63-Eu-154	723.305(5)	0.202(2)	
63-Eu-154	756.804(5)	0.0458(6)	
63-Eu-154	873.190(5)	0.1224(15)	
63-Eu-154	996.262(6)	0.1048(13)	
63-Eu-154	1004.725(7)	0.182(2)	
63-Eu-154	1274.436(6)	0.350(4)	
63-Eu-154	1494.048(9)	0.0071(2)	
63-Eu-154	1596.495(18)	0.0181(2)	
79-Au-198	411.8044(11)	0.9557(47)	(6)
80-Hg-203	279.1967(12)	0.8148(8)	(9)
83-Bi-207	569.702(2)	0.9774(3)	(5)
83-Bi-207	1063.662(4)	0.745(2)	
83-Bi-207	1770.237(9)	0.0687(4)	

Table 3 (cont)

Nuclide		Energy (keV)		Probability	Reference
90-Th-228		84.373(3)		0.0122(2)	(8)
90-Th-228	*	238.632(2)		0.435(4)	
90-Th-228	*	240.987(6)		0.0410(5)	
90-Th-228	*	277.358(10)		0.0230(3)	
90-Th-228	*	300.094(10)		0.0325(3)	
90-Th-228	*	510.77(10)	+	0.0818(10)	
90-Th-228	*	583.191(2)		0.306(2)	
90-Th-228	*	727.330(9)		0.0669(9)	
90-Th-228	*	860.564(5)		0.0450(4)	
90-Th-228	*	1620.735(10)		0.0149(5)	
90-Th-228	*	2614.533(13)		0.3586(6)	
93-Np-239		106.123(2)		0.267(4)	(10)
93-Np-239		228.183(1)		0.1112(15)	
93-Np-239		277.599(2)		0.1431(20)	
95-Am-241		26.345(1)		0.024(1)	(3)
95-Am-241		59.537(1)		0.360(4)	
95-Am-243		43.53(1)		0.0594(11)	(6)
95-Am-243		74.66(1)		0.674(10)	

* Indicates daughter in equilibrium with parent radionuclide

\+ Note the close distance to 511.003 keV annihilation radiation