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EVALUATION OF CROSS-SECTION DATA FROM THRESHOLD TO 40-60 MeV FOR SPECIFIC NEUTRON REACTIONS IMPORTANT FOR NEUTRON DOSIMETRY APPLICATIONS

Part 1

Evaluation of the excitation functions for the

$^{27}Al(n,\alpha)^{24}Na$, $^{55}Mn(n,2n)^{54}Mn$, $^{59}Co(n,p)^{59}Fe$, $^{59}Co(n,2n)^{58m+g}Co$ and $^{90}Zr(n,2n)^{89m+g}Zr$ reactions

Research Contract No 14745 R0

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Obninsk, Russia

April 2009

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Abstract

Evaluations of cross sections and their associated covariance matrices have been carried out for five dosimetry reactions:

- excitation functions were re-evaluated for the $^{27}Al(n,\alpha)^{24}Na$, $^{55}Mn(n,2n)^{54}Mn$ and $^{90}Zr(n,2n)^{89m+g}Zr$ reactions over the neutron energy range from threshold to 40 MeV;
- excitation functions were re-evaluated for the $^{59}Co(n,p)^{59}Fe$ and $^{59}Co(n,2n)^{58m+g}Co$ reactions over the neutron energy range from threshold to 60 MeV.

Uncertanties in the cross secions for all of those reactions were also derived in the form of relative covariance matrices. Benchmark calculations performed for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra show that the integral cross sections calculated from the newly evaluated excitation functions exhibit improved agreement with related experimental data when compared with the equivalent data from the IRDF-2002 library.

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1. INTRODUCTION

Cross-section data for the ^{27}Al(n,α)^{24}Na, ^{55}Mn(n,2n)^{54}Mn, ^{59}Co(n,2n)$^{58m+g}$Co, ^{59}Co(n,p)^{59}Fe and ^{90}Zr(n,2n)$^{89m+g}$Zr reactions are needed to address a wide spectrum of scientific and technical tasks. Activation detectors based on these reactions are commonly adopted in the field of reactor dosimetry. Furthermore, the ^{27}Al(n,α)^{24}Na and ^{90}Zr(n,2n)$^{89m+g}$Zr reactions are often used in experimental nuclear physics as monitor reactions for measurements of unknown cross sections by means of the activation method over the neutron energy range from 13 to 15 MeV. The ^{59}Co(n,2n)$^{58m+g}$Co reaction along with the ^{90}Zr(n,2n)$^{89m+g}$Zr reaction is also used in experimental nuclear physics for the determination of incident neutron energies.

At an IAEA Consultants' Meeting to "Review the Requirements to Improve and Extend the IRDF library (International Reactor Dosimetry File (IRDF-2002))" all of the above mentioned reactions were included in a list of proposed extensions to the IRDF-2002 database for fusion applications up to 60 MeV [1.1]. IRDF-2002 contains recommended excitation functions for the ^{27}Al(n,α)^{24}Na, ^{59}Co(n,2n)$^{58m+g}$Co and ^{90}Zr(n,2n)$^{89m+g}$Zr reactions from threshold to 20 MeV, based on data taken from the IRDF-90, version 2 library. Cross-section data for the ^{55}Mn(n,2n)^{54}Mn and ^{59}Co(n,p)^{59}Fe reactions are not included in the IRDF-2002 and JENDL/D99 files, although excitation functions for these reactions are present in ENDF/B-VII.0 from threshold to 20 MeV [1.3]. Uncertainties in these cross sections are only given for the ^{59}Co(n,p)^{59}Fe reaction. A new evaluation of the excitation funciton for the ^{55}Mn(n,2n)^{54}Mn reaction up to 150 MeV is described in Ref. [1.4]. Cross-section data for all of the reactions considered are given in the specialized MENDL-2 library from threshold to 100 MeV [1.5], but without uncertainties – the MENDL-2 library was prepared on the basis of theoretical model calculations, and is judged to be inappropriate for reactor and fusion dosimetry applications.

The main aims of this work were the re-evaluation of the cross-section data and related uncertainty covariance matrixes for the ^{27}Al(n,α)^{24}Na, ^{55}Mn(n,2n)^{54}Mn, ^{59}Co(n,2n)$^{58m+g}$Co, ^{59}Co(n,p)^{59}Fe and ^{90}Zr(n,2n)$^{89m+g}$Zr reactions and their extension to higher neutron energies up to 40 to 60 MeV. These new evaluations were performed as a consequence of improvements to the existing standards that impact on all available experimental data, coupled to consistent theoretical modelling calculations.

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2. METHOD OF EVALUATION OF THE EXCITATION FUNCTIONS FOR DOSIMETRY REACTIONS

2.1. Sources of information used in the evaluation

Two common information sources were used for the $^{27}Al(n,\alpha)^{24}Na$, $^{55}Mn(n,2n)^{54}Mn$, $^{59}Co(n,2n)^{58m+g}Co$, $^{59}Co(n,p)^{59}Fe$ and $^{90}Zr(n,2n)^{89m+g}Zr$ dosimetry reactions: differential and integral experimental data taken mainly from the EXFOR library; data and other relevant information were taken from the original publications when no records were found in EXFOR.

2.2. Analysis of experimental data

All experimental data were analyzed and, if possible, corrected with respect to the newly recommended cross-section standards for monitor reactions and recommended decay data. Corrections to the experimental data based on the new standards reduced the discrepancies, and decreased the uncertainties of the re-evaluated cross sections. The standards used to correct the microscopic experimental data under investigation are given in Table 2.1.

TABLE 2.1. DATA USED AS STANDARDS TO CORRECT THE MICROSCOPIC EXPERIMENTAL CROSS SECTIONS.

Monitor reaction	Cross section used as standard	Half-life for residual nucleus	Radiation and energy	Emission probability per decay
$^{1}H(n,n)^{1}H$	Pronyaev+ [2.1]			
$^{6}Li(n,t)^{4}He$	Pronyaev+ [2.1]			
$^{19}F(n,2n)^{18}F$	IRDF-2002 [2.2]	109.77 (5) min	Gamma 511 keV	1.9346 (8) [2.6]
$^{24}Mg(n,p)^{24}Na$	Zolotarev [2.3]	14.9590 (12) h	Gamma 1368.63 keV	1.0000(1) [2.5, 2.6]
$^{27}Al(n,\alpha)^{24}Na$	Zolotarev [*]	14.9590 (12) h	Gamma 1368.63 keV	1.0000(1) [2.5, 2.6]
$^{27}Al(n,p)^{27}Mg$	Zolotarev+ [2.4]	9.458 (12) min	Gamma 843.76 keV Gamma 1014.44 keV	0.718 (4) [2.5, 2.6] 0.280 (4) [2.5, 2.6]
$^{32}S(n,p)^{32}P$	Zolotarev [2.3]	14.263 (3) d	Beta+ 1710.48 keV	1.000 [2.6]
$^{56}Fe(n,p)^{56}Mn$	IRDF-2002 [2.2]	2.5789 (1) h	Gamma 846.754 keV Gamma 1810.72 keV	0.9887 (3) [2.5, 2.6] 0.2719 (79) [2.5, 2.6]
$^{58}Ni(n,p)^{58}Co$	IRDF-2002 [2.2]	78.86 (6) d	Gamma 511 keV Gamma 810.759 keV	0.298 (4) [2.6] 0.99450 (10) [2.6]
$^{63}Cu(n,2n)^{62}Cu$	Zolotarev [2.3]	9.73 (2) min	Beta+ 2925.8 keV Gamma 511 keV Gamma 1173.02 keV	0.9720 (2) [2.6] 1.9486 (5) [2.6] 0.00342 (5) [2.5, 2.6]
$^{65}Cu(n,2n)^{64}Cu$	Zolotarev [2.3]	12.700 (2) h	Beta+ 653.1 keV Beta– 578.7 keV Gamma 511 keV Gamma 1345.77 keV	0.1740 (22) [2.6] 0.390 (4) [2.6] 0.348 (4) [2.6] 0.00473 (10) [2.5, 2.6]
$^{64}Zn(n,p)^{64}Cu$	Zolotarev [2.3]	12.700 (2) h	Beta+ 653.1 keV Beta– 578.7 keV Gamma 511 keV Gamma 1345.77 keV	0.1740 (22) [2.6] 0.390 (4) [2.6] 0.348 (4) [2.6] 0.00473 (10) [2.5, 2.6]
$^{93}Nb(n,2n)^{92m}Nb$	IRDF-2002 [2.2]	10.15 (2) d	Gamma 934.44 keV	0.9907 (4) [2.5, 2.6]
$^{197}Au(n,2n)^{196}Au$	Zolotarev [2.3]	6.183 (10) d	Gamma 333.03 keV Gamma 355.73 keV Gamma 426.10 keV	0.229 (6) [2.5, 2.6] 0.870 (4) [2.5, 2.6] 0.066 (4) [2.5, 2.6]
$^{235}U(n,f)$	Pronyaev+ [2.1]			
$^{238}U(n,f)$	Pronyaev+ [2.1]			

Beta transitions: $E_{\beta max}$ values are listed.
[*] cross-section data from this work.

Recommended cross-section data were taken from Ref. [2.7] for the monitor reactions used in measurements of integral cross sections in ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra. Digital data for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were taken from Refs. [2.8, 2.9], respectively. Information about the isotopic compositions of the elements was obtained from Ref. [2.10].

2.3. Theoretical model calculations for the cross sections of dosimetry reactions

Theoretical model calculations provided an additional source of cross-section information for reactions with inadequate experimental data. Hence, theoretical calculations were carried out to determine the excitation functions of the $^{27}Al(n,\alpha)^{24}Na$, $^{55}Mn(n,2n)^{54}Mn$, $^{59}Co(n,2n)^{58m+g}Co$, $^{59}Co(n,p)^{59}Fe$ and $^{90}Zr(n,2n)^{89m+g}Zr$ reactions above 20 MeV.

The optical-statistical method was used for a theoretical description of the excitation function of the $^{27}Al(n,\alpha)^{24}Na$, $^{55}Mn(n,2n)^{54}Mn$, $^{59}Co(n,2n)^{58m+g}Co$, $^{59}Co(n,p)^{59}Fe$ and $^{90}Zr(n,2n)^{89m+g}Zr$ reactions, taking into account the contribution of the direct, pre-equilibrium and statistical equilibrium processes in different outgoing channels. These calculations were carried out by means of a modified version of the GNASH code [2.11, 2.12], which includes a subroutine for width fluctuation corrections.

Penetrability coefficients for neutrons were calculated on the basis of the generalized optical model, which estimates the cross sections for the direct excitations of collective low-lying levels. The ECIS code with the coupled-channel deformed optical model was used for these calculations [2.13], and the optical coefficients of the proton- and alpha-particle penetrabilities were determined by means of the SCAT2 code [2.14].

Data defining discrete level parameters for ^{27}Al, ^{55}Mn, ^{59}Co, ^{90}Zr and all residual nuclei were obtained from Ref. [2.5]. Unknown branching ratios were estimated on the basis of statistical calculations of the possible E1, E2 and M1 gamma-ray transitions. Intensities of such transitions were calculated from the radiation strength functions recommended in Ref. [2.15].

Continuum level densities were represented by means of the Gilbert-Cameron model [2.16] based on the Cook parameters [2.17] (mode IBSF = 1 in the GNASH code). Calculations of the gamma-ray transition probabilities in the continuum region of the excited states of all nuclei under consideration were made in terms of the hypothesis of the domination of the giant dipole resonance with radiative strength function from Kopecky-Uhl systematics [2.18]. Recommended parameters for the giant dipole resonances were taken from Ref. [2.19].

The modified GNASH code was used to calculate the cross sections of the $^{27}Al(n,\alpha)^{24}Na$, $^{55}Mn(n,2n)^{54}Mn$, $^{59}Co(n,2n)^{58m+g}Co$, $^{59}Co(n,p)^{59}Fe$ and $^{90}Zr(n,2n)^{89m+g}Zr$ reactions from 20 to 40-75 MeV. Data for $^{55}Mn(n,2n)^{54}Mn$ reaction were calculated from threshold to 40 MeV.

2.4. Statistical analyses of cross sections from the database

The method of statistical analysis of the correlated data was used to evaluate the excitation functions of the dosimetry reactions, as described in Refs. [2.20, 2.21]. Statistical analyses of the experimental reaction cross sections were carried out using the non-linear regression model. The following rational function was used as the model function (Pade approximation):

$$f(E) = C + \sum_{i=1}^{l_1} \frac{a_i}{E-r_i} + \sum_{k=1}^{l_2} \frac{\alpha_k (E-\varepsilon_r)+\beta_k}{(E-\varepsilon_k)^2+\gamma_k^2} ,$$

where E is the neutron energy, and C, a_i, r_i, α_k, β_k, ε_k and γ_k are the parameters to be determined. The total number of parameters of the Pade approximation is equal to $L = 2l_1 + 4l_2 + 1$.

Parameters of the model function are determined from the minimum of the functional:

$$S(\vec{\beta}) = (\vec{\sigma} - \vec{f})^T (DPD)^{-1} (\vec{\sigma} - \vec{f}),$$

in which the functional to be minimized ($\vec{\beta}$) is the vector of the parameters to be determined; $\vec{\sigma}$ is the vector of cross sections from the database; D is the diagonal matrix of the uncertainty of the cross sections from the database; P is the correlation matrix of the experimental data used to evaluate the excitation function; and the superscript T denotes a transpose.

Technical aspects of the minimization process based on the use of the discrete optimization method and Newton-Gauss algorithm are described in Ref. [2.22]. The algorithm used to minimize $S(\vec{\beta})$ contains two approximations that simplify the calculation appreciably:

1) cross-section data obtained in different experiments are assumed to be uncorrelated;
2) an average correlation coefficient is used to describe the correlations between cross sections measured in one experiment.

The covariance matrix of the uncertainties of the evaluated parameters $W(\vec{\beta})$ and the uncertainties of the evaluated function at point $\Delta f(E^k_{i_k}, \vec{\beta})$ are determined from the relationships:

$$W(\vec{\beta}) = \tfrac{s}{n-L} (X^T V^{-1} X)^{-1},$$

$$\Delta f(E_{i_k}, \vec{\beta}) = \sum_{m=1}^{L} \sum_{j=1}^{L} X^k_{i_k m} X^k_{i_k j} W_{mj},$$

where n is the total number of cross-section data used in the analysis of a reaction, and X is the $(n \times L)$ matrix of the coefficients of sensitivity of the rational function to a change in parameters based on:

$$X_{i_k m} = \frac{\partial f(E_{i_k}, \vec{\beta})}{\partial \beta_m}.$$

The structure of the uncertainties for all experimental data was analyzed to determine the average correlation coefficients. The average correlation coefficient $\vec{p}^k$ for the k^{th} experiment containing information on the n_k values of the reaction excitation function was determined by means of the formulae:

$$\vec{p}^k = \frac{2}{(n_k - 1) n_k} \sum_{i=1}^{n_k - 1} \sum_{j=i+1}^{n_k} \frac{\sum_{m=1}^{l} P^m_{ij} e^m_i e^m_j}{e_i e_j},$$

where $e_i(e_j)$ is the total uncertainty (standard deviation) of the cross section at the i^{th} (j^{th}) point corresponding to a standard deviation of 1σ; e^m_i (e^m_j) is the m^{th} component of the systematic uncertainty of the cross section at the i^{th} (j^{th}) point; P^m_{ij} is the coefficient of the correlation between the m^{th} components of the systematic uncertainties at the i^{th} (j^{th}) points; and l is the number of components of the systematic uncertainty. This method of statistical analysis of the correlated data was performed by means of the PADE-2 code [2.20].

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3. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE $^{27}Al(n,\alpha)^{24}Na$ REACTION

The ^{27}Al isotopic abundance in natural aluminium is 100 atom percent, and the ^{24}Na obtained via the (n,α) reaction undergoes 100% β^- decay with a half-life of (14.9590 ± 0.0012) hours. 1368.633-keV gamma radiation (I_γ = 1.0000 ± 0.0001) and 2754.028-keV gamma radiation (I_γ = 0.99944 ± 0.00004) are normally used to determine the $^{24}Al(n,\alpha)^{24}Na$ reaction rate. Recommended decay data for the half-life and gamma-ray emission probabilities per decay of ^{24}Na were taken from Ref. [2.6] of Section 2.

Microscopic experimental data were analyzed during the preparation of the assembled input database in order to evaluate the cross sections and uncertainties for the $^{24}Al(n,\alpha)^{24}Na$ reaction [3.1-3.72]. During this procedure, various experimental data were corrected [3.2, 3.4, 3.6-3.8, 3.10, 3.11, 3.13, 3.15-3.18, 3.20-3.22, 3.27-3.29, 3.32, 3.34, 3.35, 3.39-3.41, 3.43, 3.44, 3.46, 3.48, 3.49, 3.54, 3.56] on the basis of the newly recommended cross-section data for the relevant monitor reactions and the recommended decay data (see Table 2.1). Other corrections were also applied to some of the experimental data of Refs. [3.5, 3.9, 3.16, 3.19, 3.21, 3.33, 3.43]. Cross-section data measured by Tewes *et al.* [3.5] for incident neutron energies of 8.4 to 14.0 MeV were renormalized to the recent experimental data of Mannhart and Schmidt [3.56]. A correction factor of Fc = 1.36146, was determined from the ratio of cross-section integrals in the overlapping energy region from 8.4 to 14.0 MeV.

The experimental data of Tewes *et al.* [3.5], Gabbard and Kern [3.9], Paulsen and Liskien [3.16] obtained in measurements with $T(d,n)^4He$ neutron sources, and all the experimental data of Menlove *et al.* [3.21] were renormalized by factors Fc = 1.10138, 0.99177, 0.96830 and 1.07669, respectively. Correction factors were determined from the ratio of cross-section integrals of Ikeda *et al.* [3.50] and Filatenkov *et al.* [3.54] to adequate integrals for these experimental data. Data of Hemingway *et al.* [3.19] and Welch *et al.* [3.33] were renormalized to values of 110.6 mb at 14.8 MeV and 38.51 mb at 20 MeV, respectively. Janczyszyn *et al.* measured the cross-section ratio of $^{24}Mg(n,p)^{24}Na$ to $^{27}Al(n,\alpha)^{24}Na$ reactions in the energy range 13.59 to 17.86 MeV [3.43]. The cross sections of the $^{27}Al(n,\alpha)^{24}Na$ reaction were determined from the reverse ratios by using data for the $^{24}Mg(n,p)^{24}Na$ excitation function from Ref. [2.2]. Experimental data of Janczyszyn *et al.* were obtained from measurements on Mg samples of natural isotopic composition – these data was corrected for contributions from the $^{25}Mg(n,x)^{24}Na$ and $^{26}Mg(n,t)^{24}Na$ reactions (Ref. [3.73]). Tsabaris *et al.* [3.52] registered the outgoing α-particles from an Al target at incident neutron energies of 6.28-, 8.0- and 9.0-MeV to derive well-defined cross sections for the $^{27}Al(n,\alpha)^{24}Na$ reaction.

Cross-section data from Refs. [3.57-3.71] was rejected due to their significant deviation from the main bulk of experimental data. Within these rejected data, the cross-section values reported in Refs. [3.57, 3.59-3.61, 3.64, 3.65] comprised only one or two energy points from 14 to 15 MeV.

The excitation function for the $^{27}Al(n,\alpha)^{24}Na$ reaction in the energy region from threshold to 40 MeV was evaluated by means of statistical analyses of the experimental cross-section data [3.1-3.56] and data obtained from theoretical model calculations. Above a neutron energy of 23.8 MeV, evaluated data are total cross sections of the $^{27}Al(n,\alpha)^{24}Na$, $^{27}Al(n,p+^3H)^{24}Na$ (E_{th} = 23.804 MeV), $^{27}Al(n,n+^3He)^{24}Na$ (E_{th} = 24.596 MeV), $^{27}Al(n,d+d)^{24}Na$ (E_{th} = 27.987 MeV), $^{27}Al(n,n+p+d)^{24}Na$ (E_{th} = 30.295 MeV) and $^{27}Al(n,2n+2p)^{24}Na$ (E_{th} = 32.603 MeV) reactions.

Uncertainties in the evaluated excitation function for the ^{27}Al(n,α+)^{24}Na reaction are given in the form of a relative covariance matrix for 49-neutron energy groups (LB = 5). Covariance matrix uncertainties were calculated simultaneously with the recommended cross-section data by means of the PADE-2 code.

Six-digit eigenvalues for the relative covariance matrix in File-33 are as follows:

1.47957E-07	1.53175E-07	1.59656E-07	1.65987E-07
1.71756E-07	1.77453E-07	1.82693E-07	1.88689E-07
1.96425E-07	2.04939E-07	2.16677E-07	2.30205E-07
2.44356E-07	2.63615E-07	2.86473E-07	3.08552E-07
3.42955E-07	3.77898E-07	4.20954E-07	4.79727E-07
5.36077E-07	6.26585E-07	7.09177E-07	8.37629E-07
9.76062E-07	1.14156E-06	1.39579E-06	1.73935E-06
2.57266E-06	4.15580E-06	6.49984E-06	1.99568E-05
3.09736E-05	5.40791E-05	6.60122E-05	8.86863E-05
1.49013E-04	1.90158E-04	1.96350E-04	2.34926E-04
2.80484E-04	3.31398E-04	5.34263E-04	6.22076E-04
1.23747E-03	1.40976E-03	6.94189E-03	3.12400E-02
4.19312E-02			

Evaluated group cross sections and their uncertainties for the excitation function of the ^{27}Al(n,α+)^{24}Na reaction are listed in Table 3.1. Group boundaries are the same as in File-33.

As shown in Table 3.1, the smallest uncertainties in the evaluated cross sections of 0.37% to 0.49% are observed in the neutron energy range from 13.75 to 15.0 MeV. Uncertainties lower than 1% are also observed in the neutron energy ranges 7.75 to 12.00, 13.00 to 13.50 and 15.00 to 18.50 MeV. A significant uncertainty of 20.3% in the cross sections from threshold to 6.0 MeV arises from the large uncertainties in the experimental data within this region and the existing discrepancies between experimental data. Inadequate experimental information above 25 MeV results in the uncertainties of the evaluated cross sections increasing from 5.45% to 14.75%.

Fig. 3.1 compares the re-evaluated excitation function for the ^{27}Al(n,α)^{24}Na reaction over the neutron energy range from threshold to 40.0 MeV with IRDF-2002, ENDF/B-VII.0, MENDL-2 and experimental data obtained between 1957 and 1975. Comparison of the evaluated excitation functions with experimental data obtained from 1975 to 2007 is shown in Fig. 3.2. The same evaluated excitation functions and rejected experimental data are presented in Fig. 3.3.

Integral experiments for the ^{27}Al(n,α)^{24}Na reaction are described in Refs. [3.74-3.88]. Twelve experiments was carried out in neutron fields with similar spectra to the ^{235}U thermal fission neutron spectrum [3.74-3.85], and three experiments were performed in a ^{252}Cf spontaneous fission neutron spectrum [3.86-3.88]. Experimental data obtained for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were corrected with respect to the newly recommended cross sections for the monitor reactions and decay data.

Measured integral cross sections for the ^{252}Cf spontaneous fission neutron spectrum [3.86-3.88] range from (0.86 ± 0.05) mb [3.86] to (1.048 ± 0.051) mb [3.87], while a value of (1.006 ± 0.022) mb has been obtained by Mannhart and Alberts [3.88].

Measurements of the integral cross sections for the ^{235}U thermal fission neutron spectrum range from 0.500 to 0.780 mb [3.74-3.85]. The lowest value of 0.500 mb was obtained by Shikata in studies with the JRR-1 reactor [3.76], although no information on the uncertainty is given in this publication, while a value of (0.780 ± 0.030) mb was measured by Fabry [3.79].

TABLE 3.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{27}Al(n,\alpha+)^{24}Na$ REACTION IN THE ENERGY RANGE FROM THRESHOLD TO 40 MeV.

Neutron energy (MeV) from – to	Cross-section (mb)	Uncertainty (%)	Neutron energy (MeV) from – to	Cross-section (mb)	Uncertainty (%)
3.249 – 6.000	0.157	20.30	12.500 – 13.000	122.189	1.04
6.000 – 6.500	3.279	2.05	13.000 – 13.500	124.952	0.80
6.500 – 6.750	8.188	1.77	13.500 – 13.750	125.079	0.52
6.750 – 7.000	13.554	1.66	13.750 – 14.000	123.424	0.49
7.000 – 7.250	19.142	1.45	14.000 – 14.200	120.971	0.46
7.250 – 7.500	24.599	1.17	14.200 – 14.400	118.321	0.41
7.500 – 7.750	30.382	1.00	14.400 – 14.600	115.478	0.37
7.750 – 8.000	36.732	0.93	14.600 – 14.800	112.559	0.38
8.000 – 8.250	43.575	0.88	14.800 – 15.000	109.601	0.44
8.250 – 8.500	50.711	0.86	15.000 – 15.500	104.313	0.55
8.500 – 8.750	57.893	0.86	15.500 – 16.000	96.504	0.67
8.750 – 9.000	64.889	0.87	16.000 – 16.500	88.438	0.73
9.000 – 9.250	71.519	0.87	16.500 – 17.000	80.317	0.77
9.250 – 9.500	77.667	0.88	17.000 – 17.500	72.387	0.83
9.500 – 9.750	83.281	0.89	17.500 – 18.000	64.852	0.90
9.750 – 10.000	88.359	0.91	18.000 – 18.500	57.855	0.98
10.000 – 10.250	92.933	0.93	18.500 – 19.000	51.471	1.08
10.250 – 10.500	97.054	0.95	19.000 – 19.500	45.726	1.19
10.500 – 10.750	100.777	0.97	19.500 – 20.000	40.605	1.31
10.750 – 11.000	104.155	0.97	20.000 – 22.500	28.990	1.71
11.000 – 11.250	107.237	0.97	22.500 – 25.000	16.715	2.81
11.250 – 11.500	110.060	0.98	25.000 – 30.000	8.226	5.45
11.500 – 11.750	112.658	0.98	30.000 – 35.000	3.646	11.98
11.750 – 12.000	115.059	0.99	35.000 – 40.000	3.056	14.75
12.000 – 12.500	118.324	1.02			

Neutron spectra measurements show that the standard ^{235}U thermal fission neutron spectrum may be obtained from a thermal column with 90%-enriched ^{235}U fission plate. Experimental data obtained from measurements in reactor cores and critical assemblies need to be corrected for differences between the real spectrum and the standard ^{235}U thermal fission neutron spectrum. Determination of this adjustment factor is a significant problem, and represents the major source of uncertainty in the resulting cross section. The more representative experimental data measured in facilities with an enriched ^{235}U fission converter were obtained by Boldeman (0.691 ± 0.035) mb [3.75]. More recent experimental studies by Grigor'ev *et al.* (0.690 ± 0.025) mb [3.83], Mannhart (0.7008 ± 0.0221) mb [3.84] and Arribere *et al.* (0.7028 ± 0.0216) mb [3.85] agree within the limits of uncertainty of the Boldeman data.

Mannhart has analyzed the experimental data for both spectra with great care, and obtained values of (0.7007 ± 0.0090) mb for the ^{235}U thermal fission neutron spectrum and (1.016 ± 0.013) mb for the ^{252}Cf spontaneous fission neutron spectrum [3.89].

All of the evaluated experimental data were used in benchmark calculations. The results of these tests with the re-evaluated excitation function of the $^{27}Al(n,\alpha)^{24}Na$ reaction are given in Table 3.2, in which C/E is the ratio of the calculated to experimental cross section. These data show that the average cross sections calculated from the re-evaluated excitation function for the $^{27}Al(n,\alpha)^{24}Na$ reaction exhibit greater agreement with the integral experimental data than the equivalent data from the IRDF-2002 and ENDF/B-VII.0 libraries. Extremely high discrepancies are observed between the calculated and experimental data of the MENDL-2 library.

TABLE 3.2. CALCULATED AND MEASURED AVERAGE CROSS SECTIONS FOR THE $^{27}Al(n,\alpha)^{24}Na$ REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	Average cross section, mb		C/E
	Calculated	Measured	
^{235}U thermal fission neutron spectrum	0.71373 [A]	0.7007 ± 0.0090 [3.89]	1.01860
	0.72718 [B]		1.03779
	0.72739 [C]		1.03809
	0.40838 [D]		0.58282
^{252}Cf spontaneous fission neutron spectrum	1.0182 [A]	1.016 ± 0.013 [3.89]	1.00217
	1.0369 [B]		1.02057
	1.0366 [C]		1.02028
	0.60648 [D]		0.59693

[A] present evaluation.
[B] IRDF-2002 (IRDF-90 version 2).
[C] ENDF/B-VII.0 (ENDF/B-VI revision 8).
[D] MENDL-2.

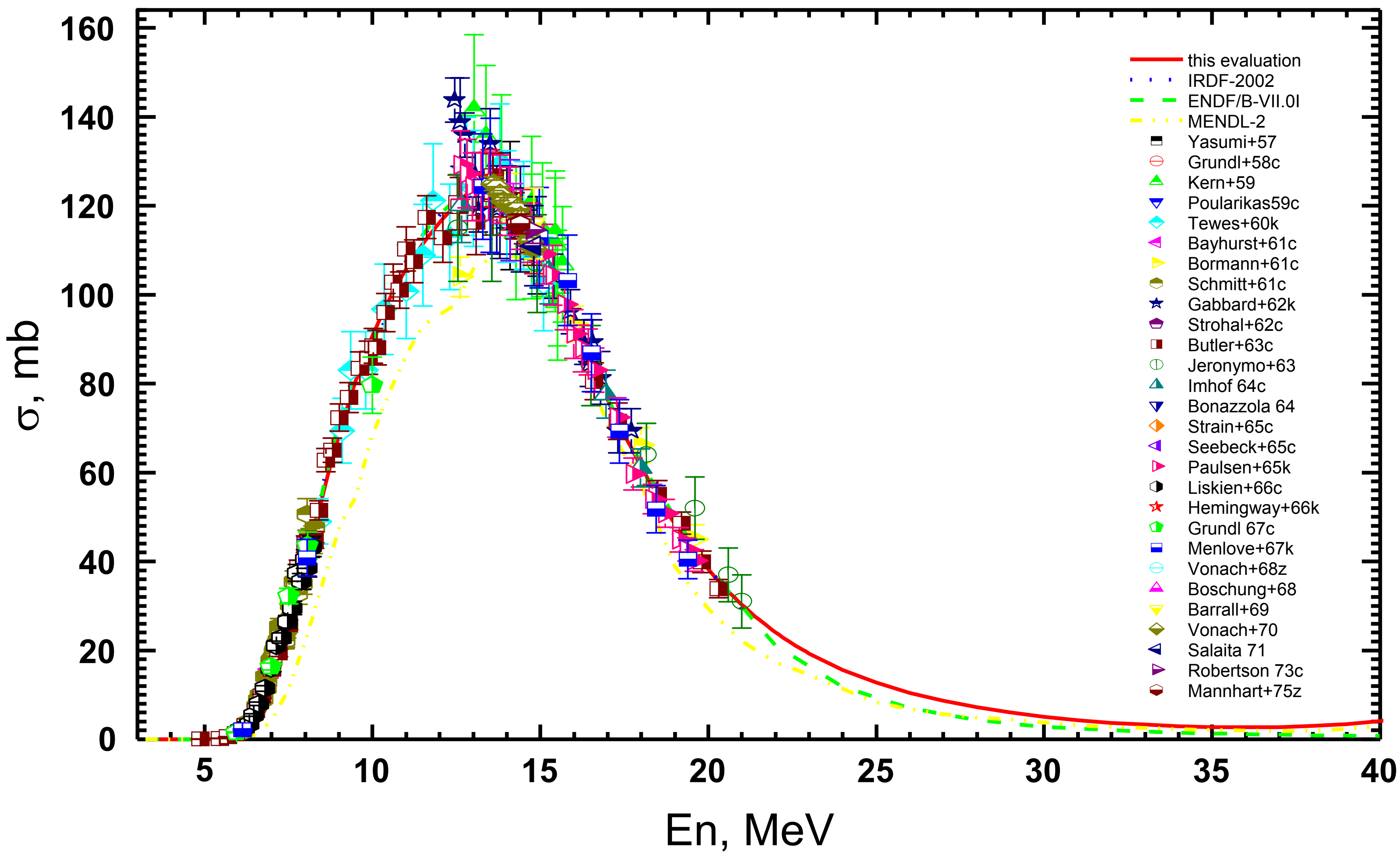

FIG. 3.1. Re-evaluated excitation function of the $^{27}Al(n,\alpha)^{24}Na$ reaction in the energy range from threshold to 40 MeV in comparison with IRDF-2002, ENDF/B-VII.0, MENDL-2 and experimental data from 1957 to 1975.

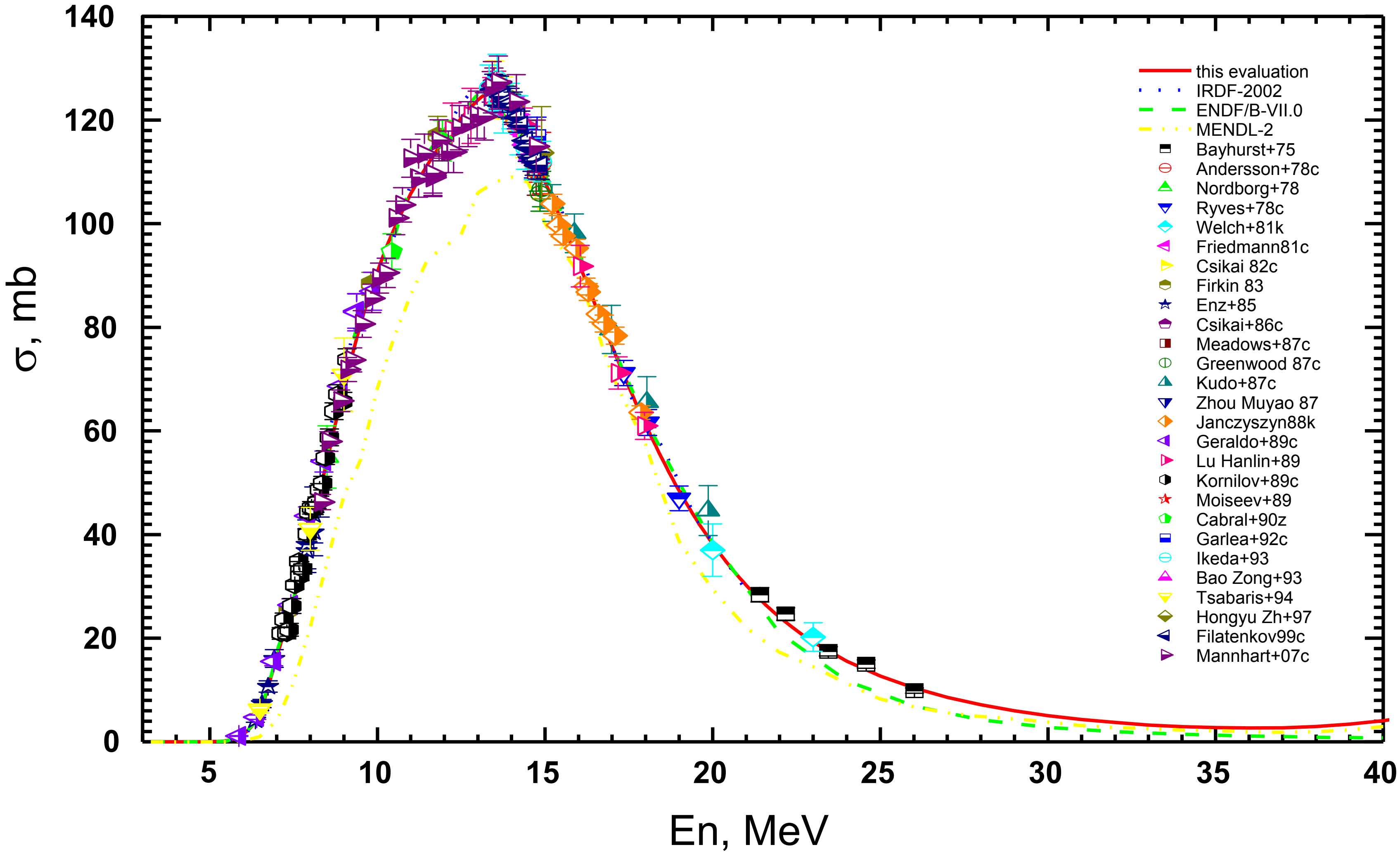

FIG. 3.2. Re-evaluated excitation function of the $^{27}Al(n,\alpha)^{24}Na$ reaction in the energy range from threshold to 40 MeV in comparison with IRDF-2002, ENDF/B-VII.0, MENDL-2 and experimental data from 1975 to 2007.

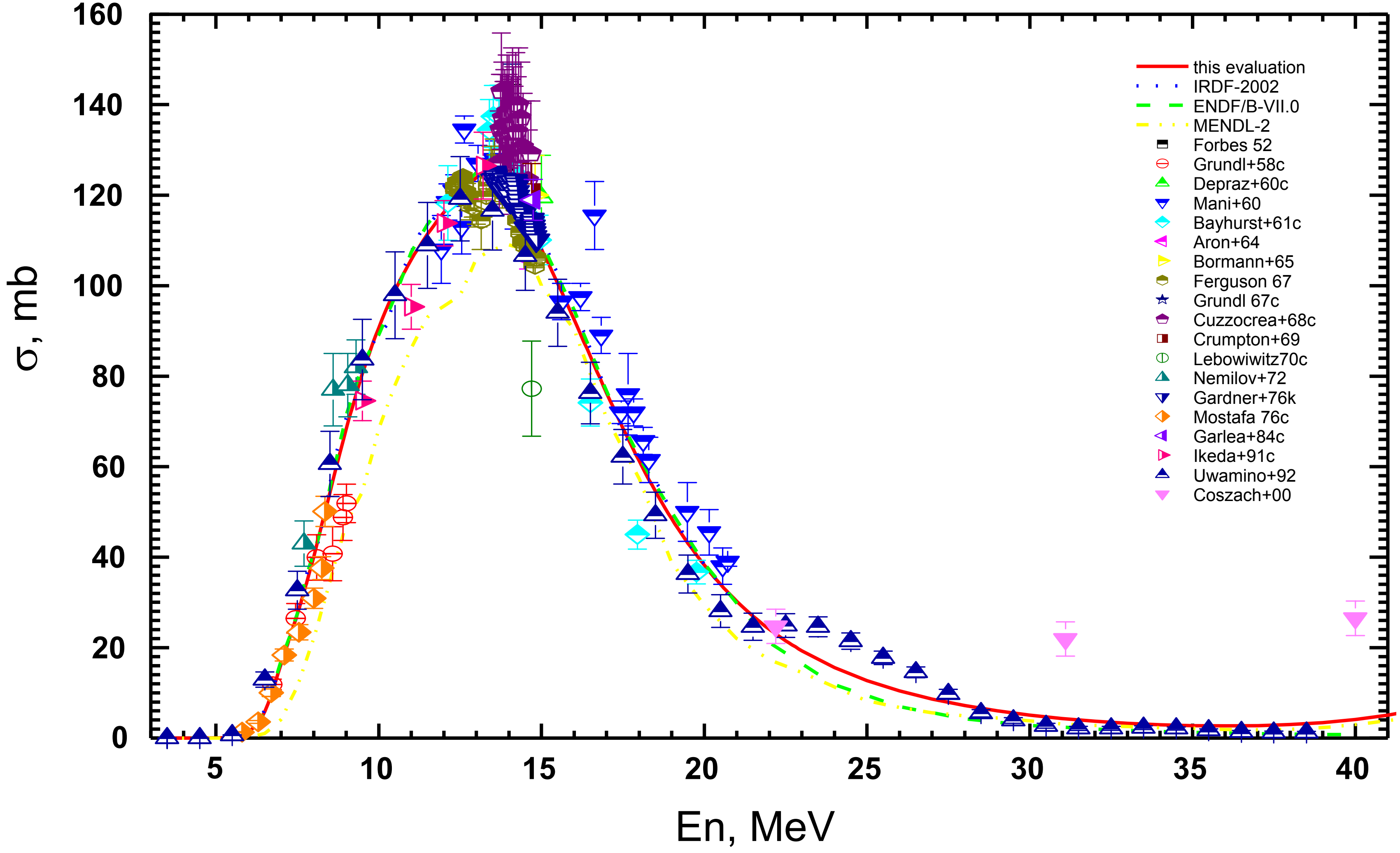

FIG. 3.3. Re-evaluated excitation function of the $^{27}Al(n,\alpha)^{24}Na$ reaction in the energy range from threshold to 40 MeV in comparison with IRDF-2002, ENDF/B-VII.0, MENDL-2 and rejected experimental data.

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4. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE ^{55}Mn(n,2n)^{54}Mn REACTION

The isotopic abundance of ^{55}Mn in natural manganese is 100 atom percent, and the ^{54}Mn obtained via the (n,2n) reaction undergoes 100% EC decay with a half-life of (312.12 ± 0.10) days. 834.848-keV gamma radiation (I_γ = 0.99976 ± 0.00001) is normally used to determine the ^{55}Mn(n,2n)^{54}Mn reaction rate. Recommended decay data for the half-life and gamma-ray emission probability per decay of ^{54}Mn were taken from Ref. [2.6] of Section 2.

Microscopic experimental data were analyzed during the preparation of the assembled input database in order to evaluate the cross sections and uncertainties for the ^{55}Mn(n,2n)^{54}Mn reaction [4.1-4.33]. Various experimental data were corrected [4.1-4.4, 4.6, 4.8-4.12, 4.14-4.16, 4.18, 4.19, 4.22-4.24, 4.29, 4.32] on the basis of the newly recommended cross-section data for the relevant monitor reactions and the recommended decay data (see Table 2.1).

Specific adjustments were applied to some of the experimental data as outlined below. Cross sections measured by Filatenkov and Chuvaev in the neutron energies interval 14.42 to 14.78 meV [4.24] were used as a reference data for the correction of experimental data from Refs. [4.3 and 4.17]. After corrections to the new standards experimental data of Pausen and Liskien [4.3] and Zhao Wenrong *et al.* [4.17] were renormalized by factors of Fc = 0.81319 and 0.95100, respectively. Neutron energies of 11.14, 11.97 and 12.85 MeV as reported by Bostan and Qaim [4.22] were shifted by + 0.45 MeV. Cross sections measured by Soewarsono *et al.* [4.20] over a wide neutron energy range of 17.55 to 38.5 MeV were renormalized to a value of 820 mb at 17.55 MeV, as derived from the experimental data of Uwamino *et al.* [4.21].

The database used to evaluate the excitation function of the ^{55}Mn(n,2n)^{54}Mn reaction was assembled from microscopic experimental data [4.1-4.24] and theoretical modelling calculations. Cross sections determined in Refs. [4.25-4.33] were rejected due to their significant overestimation or underestimation of the cross sections of the ^{55}Mn(n,2n)^{54}Mn reaction, containing only one experimental value in the energy range from 14 to 15 MeV.

Evaluation of the excitation function of the ^{55}Mn(n,2n)^{54}Mn reaction from threshold to 40 MeV was carried out by means of the generalized least-squares method within the PADE-2 code. Uncertainties in the evaluated excitation function for the ^{55}Mn(n,2n)^{54}Mn reaction are given in the form of a relative covariance matrix for 39-neutron energy groups (LB = 5). Covariance matrix uncertainties were calculated simultaneously with the recommended cross-section data by means of the PADE-2 code.

Six-digit eigenvalues for the relative covariance matrix in File-33 are as follows:

9.60189E-06	9.64312E-06	9.73051E-06	9.85035E-06
1.00065E-05	1.02262E-05	1.04988E-05	1.07979E-05
1.11916E-05	1.18366E-05	1.26678E-05	1.36336E-05
1.58284E-05	1.75566E-05	2.24423E-05	2.62145E-05
3.37354E-05	4.71224E-05	6.03072E-05	7.19641E-05
9.68428E-05	1.30367E-04	1.71642E-04	2.20387E-04
2.75154E-04	3.25769E-04	3.63859E-04	4.27408E-04
4.37291E-04	5.07300E-04	5.98776E-04	1.01930E-03
2.45165E-03	3.58243E-03	6.03353E-03	6.07530E-02
8.38378E-02	2.25620E-01	3.56662E-01	

Evaluated group cross sections and the uncertainties of the excitation function for the $^{55}Mn(n,2n)^{54}Mn$ reaction are listed in Table 4.1. Group boundaries are the same as in File-33.

TABLE 4.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{55}Mn(n,2n)^{54}Mn$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 40 MeV

Neutron energy (MeV) from – to	Cross section (mb)	Uncer-Tainty (%)	Neutron energy (MeV) from – to	Cross section (mb)	Uncer-Tainty (%)
10.414 – 11.000	10.743	26.11	21.000 – 22.000	809.870	5.08
11.000 – 11.500	74.006	7.24	22.000 – 23.000	752.537	8.04
11.500 – 12.000	191.891	4.26	23.000 – 24.000	678.816	10.94
12.000 – 12.500	338.448	3.14	24.000 – 25.000	597.319	13.17
12.500 – 13.000	478.568	2.36	25.000 – 26.000	516.646	14.65
13.000 – 13.500	590.866	1.62	26.000 – 27.000	442.846	15.67
13.500 – 14.000	671.588	1.25	27.000 – 28.000	378.836	16.53
14.000 – 14.500	726.508	1.09	28.000 – 29.000	325.139	17.32
14.500 – 15.000	763.398	1.07	29.000 – 30.000	280.932	17.94
15.000 – 15.500	788.720	1.22	30.000 – 31.000	244.845	18.27
15.500 – 16.000	806.978	1.45	31.000 – 32.000	215.434	18.28
16.000 – 16.500	821.019	1.64	32.000 – 33.000	191.395	18.09
16.500 – 17.000	832.485	1.73	33.000 – 34.000	171.633	17.97
17.000 – 17.500	842.180	1.77	34.000 – 35.000	155.271	18.26
17.500 – 18.000	850.319	1.79	35.000 – 36.000	141.610	19.31
18.000 – 18.500	856.700	1.83	36.000 – 37.000	130.107	21.28
18.500 – 19.000	860.824	1.91	37.000 – 38.000	120.338	24.16
19.000 – 19.500	861.992	2.02	38.000 – 39.000	111.971	27.79
19.500 – 20.000	859.407	2.21	39.000 – 40.000	104.745	31.96
20.000 – 21.000	845.701	2.93			

Table 4.1 reveals that the smallest uncertainties in the evaluated cross sections of 1.07% to 1.09% are observed in the neutron energy range from 14.0 to 15.0 MeV, while these uncertainties are highest near the threshold and above 21 MeV.

Fig. 4.1 compares the re-evaluated excitation function for the $^{55}Mn(n,2n)^{54}Mn$ reaction over the neutron energy range from threshold to 40.0 MeV with the equivalent Karlruhe-2007 and ENDF/B-VII.0 excitation functions and experimental data. These same evaluated cross sections and rejected experimental data are shown in Fig. 4.2. Comparison of the excitation functions shows that ENDF/B-VII.0 and these evaluations agree well in the energy range 13 to 17 MeV. Below 13 MeV, the ENDF/B-VII.0 evaluation recommends systematically higher cross sections, while above 17 MeV cross sections are systematically lower than those for this evaluation. The re-evaluated excitation function and Karlsruhe-2007 data are in good agreement over the energy range from 20 to 35 MeV. MENDL-2 library contains an excitation function for the $^{55}Mn(n,2n)^{54}Mn$ reaction that overestimates the cross sections systematically, especially for neutron energies between 12 and 23 MeV.

Integral experimental data for the $^{55}Mn(n,2n)^{54}Mn$ reaction are given in Refs. [4.34-4.42]. Seven experiments were carried out in neutron fields with similar spectra to the ^{235}U thermal fission neutron spectrum [4.34-4.40], while two studies were performed in a ^{252}Cf spontaneous fission neutron spectrum [4.41, 4.42].

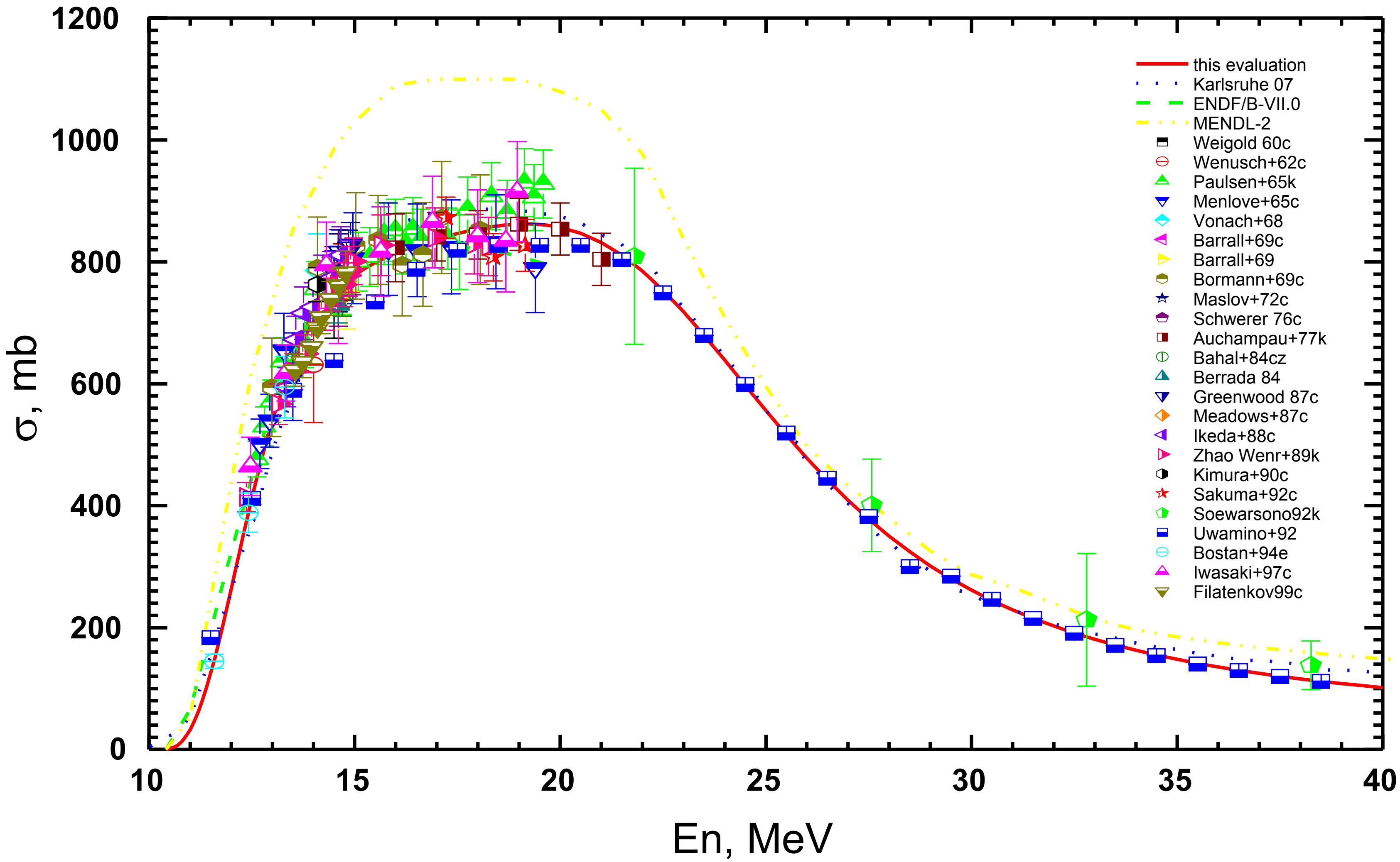

FIG. 4.1. Re-evaluated excitation function of the $^{55}Mn(n,2n)^{54}Mn$ reaction in the energy range from threshold to 40 MeV in comparison with Karlsruhe-2007, ENDF/B-VII.0, MENDL-2 and experimental data.

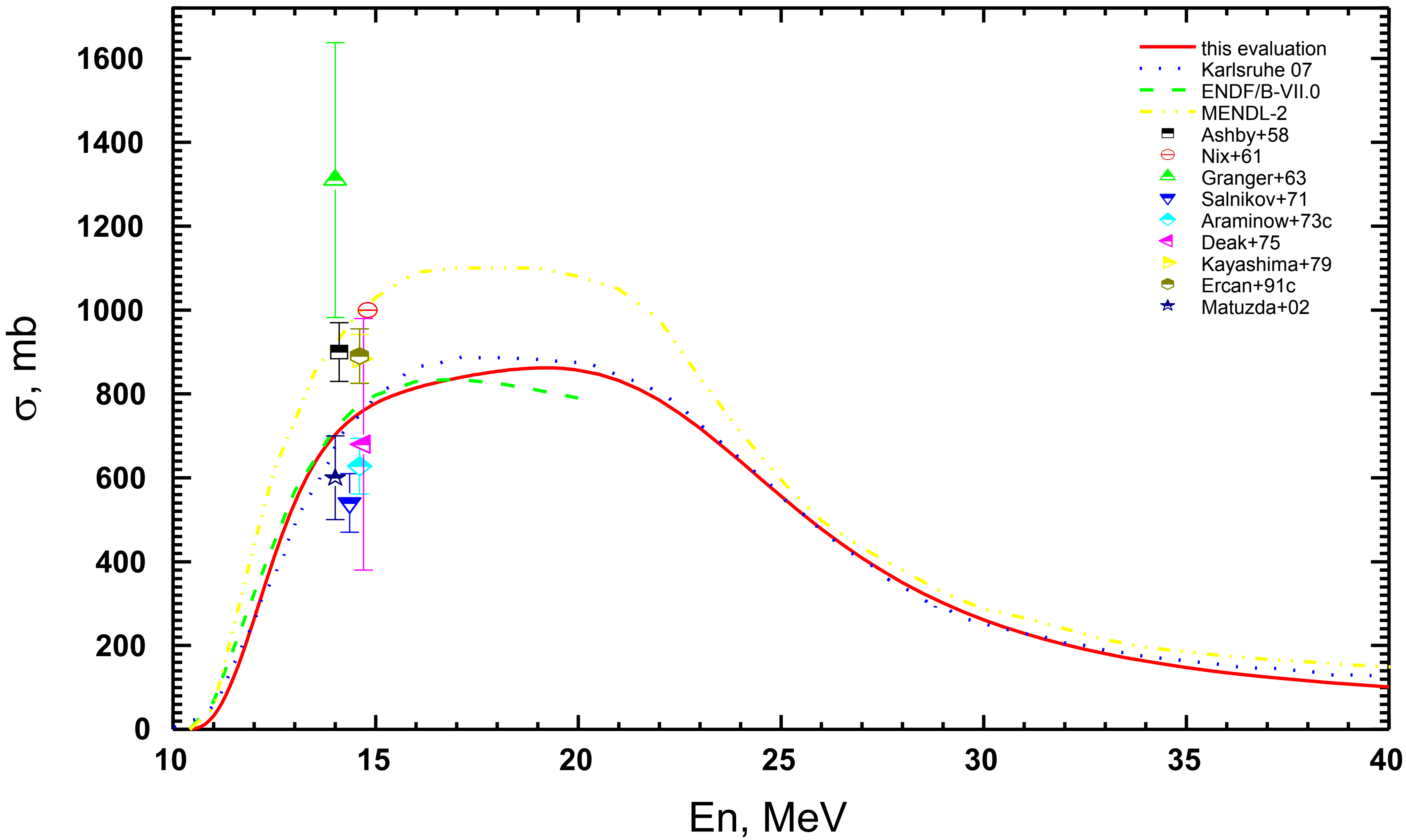

FIG 4.2. Re-evaluated excitation function of the $^{55}Mn(n,2n)^{54}Mn$ reaction in the energy range from threshold to 40 MeV in comparison with Karlsruhe-2007, ENDF/B-VII.0, MENDL-2 and rejected experimental data.

Measured integral cross sections for the ^{235}U thermal fission neutron spectrum extend over 0.2020 to 0.2410 mb [4.34-4.40]. The more representative measurements were carried out by Fabri and DeWorm (0.207 ± 0.011) mb [4.34] and Kobayashi and Kimura (0.202 ± 0.010) mb [4.38] in which a ^{235}U fission spectrum was generated by an enriched ^{235}U fission plate converter. An average-weighted value of (0.2043 ± 0.0074) mb was obtained frm these experimental data.

Measured integral cross sections for the ^{252}Cf spontaneous fission neutron spectrum range from (0.4104 ± 0.0091) [4.42] to (0.580 ± 0.140) mb [4.41]. Experimental data [4.42] agree well with the evaluated cross section of (0.4075 ± 2.53) mb [4.43]. The measurements of Dezsö and Csikai give a significantly higher value of (0.580 ± 0.140) mb [4.41] that was not taken into account in the benchmark calculations.

Evaluated excitation functions for the $^{55}Mn(n,2n)^{54}Mn$ reaction were tested against the representative integral experimental data. Calculated average cross sections for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are compared with the ENDF/B-VII.0, Karlsruhe-2007, MENDL-2 and experimental data in Table 4.2.

TABLE 4.2. CALCULATED AND MEASURED AVERAGED CROSS SECTIONS FOR THE $^{55}Mn(n,2n)^{54}Mn$ REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	**Average cross section, mb**		**C/E [4.43]**
	Calculated	**Measured**	
^{235}U thermal fission neutron spectrum	0.20410 [A]	0.2043 ± 0.0740 [*]	0.99902
	0.24159 [B]		1.18253
	0.21714 [C]		1.06285
	0.30852 [D]		1.51013
^{252}Cf spontaneous fission neutron spectrum	0.41615 [A]	0.4075 ± 0.0095 [4.43]	1.02123
	0.48127 [B]		1.18103
	0.43503 [C]		1.06756
	0.61770 [D]		1.51583

[A] present evaluation.
[B] ENDF/B-VII.0.
[C] Karlsruhe-2007.
[D] MENDL-2.
[*] average-weighted value obtained from experimental data [4.34, 4.38].

These data show that the average cross sections calculated from the re-evaluated excitation function for ^{235}U thermal fission neutron spectrum and ^{252}Cf spontaneous fission neutron spectrum agree well with equivalent experimental data. Discrepancies between the ENDF/B-VII.0 and experimental data are 18.3% and 18.1% for the ^{235}U and ^{252}Cf spectra, respectively. The Karlsruhe-2007 evaluation exhibits better agreement with integral experimental data than the ENDF/B-VII.0 evaluation. Discrepancies between the Karlsruhe-2007 and experimental data are 6.3% and 6.8% for the ^{235}U and ^{252}Cf spectra, respectively. Average cross sections calculated from MENDL-2 exceed experimental data by approximately 51%.

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5. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE $^{59}Co(n,p)^{59}Fe$ REACTION

The isotopic abundance of ^{59}Co in natural cobalt is 100 atom percent, and the ^{59}Fe obtained via the (n,p) reaction undergoes 100% β^- decay with a half-life of (44.495 ± 0.009) days. Two of the most intensive gamma rays at 1099.245 keV ($I_\gamma = 0.565 \pm 0.018$) and 1291.590 keV ($I_\gamma = 0.432 \pm 0.014$) are normally used to determine the $^{59}Co(n,p)^{59}Fe$ reaction rate. Recommended decay data for the half-life, and beta and gamma-ray emission probabilities per decay of ^{59}Fe were taken from Ref. [2.6] of Section 2.

Microscopic experimental data were analyzed during the preparation of the assembled input database in order to evaluate the cross sections and uncertainties for the $^{59}Co(n,p)^{59}Fe$ reaction [5.1-5.34]. During this procedure, various experimental data were corrected [5.3-5.18, 5.21-5.26, 5.30-5.34] on the basis of the newly recommended cross-section data for the relevant monitor reactions and the recommended decay data (see Table 2.1). Other adjustments were also applied to some of the experimental data of Refs. [5.2, 5.3, 5.7, 5.16, 5.18]. Thus, the data of Smith and Meadows obtained with neutrons from a $^{7}Li(p,n)^{7}Be$ source [5.3] in the energy range of 4.334 to 5.536 MeV were renormalized to more recent experimental data with this same source [5.4]. Data from these authors obtained with a $D(d,n)^{3}He$ neutron source were also renormalized to Mannhart and Schmidt measurements [5.24] in the overlapping energy range of 5.737 to 9.944 MeV. After corrections with respect to the new standards, the data from Ref. [5.3] were multiplied by factors of Fc = 0.92332 and 1.13713, respectively.

Cross sections measured by Ikeda *et al.* in the neutron energy range from 13.32 to 14.91 MeV [5.12] and Filatenkov and Chuvaev from 13.56 to 14.78 MeV [5.21] were used as reference data in the correction of experimental data from Refs. [5.16, 5.18]. After corrections with respect to the new standards, the experimental data of Viennot *et al.* [5.16] and Molla *et al.* [5.18] were renormalized by factors of Fc = 1.09838 and 0.92926, respectively.

The results of measurements by Jeronymo *et al.* over a wide energy range from 12.55 to 20.6 MeV [5.2] were renormalized to a value of (45.27 ± 1.97) mb at 14.90 MeV as obtained by Greenwood [5.11].

Data reported by Williams *et al.* [5.7] were multiplied by factor of Fc = 0.93113 derived from the new standards. This value was determined from two ratios:

1. ratio of the cross-section integrals of Li Tingyan *et al.* [5.14] and Williams *et al.* [5.7] over the energy range of 14.20 to 14.77 MeV, R = 0.92182, and
2. ratio of the cross-section integrals of Semkova *et al.* [5.23] and Williams *et al.* [5.7] over the energy range of 14.83 to 18.20 MeV, R = 0.94043.

Cross-section data given in Refs. [5.25-5.34] were rejected due to their large deviations from a significant amount of the other extensive experimental data. Furthermore, apart from the measurements of Cezar Suita *et al.* [5.33], these rejected experimental data had only been measured at one energy point between 14 and 15 MeV.

The excitation function for the $^{59}Co(n,p)^{59}Fe$ reaction in the energy range from threshold to 60 MeV was evaluated by means of statistical analyses of the experimental cross-section data [5.1-5.24] and theoretical modelling calculations undertaken by means of the PADE-2 code. Uncertainties in the evaluated excitation function for the $^{59}Co(n,p)^{59}Fe$ reaction are given in the form of a relative covariance matrix for 49-neutron energy groups (LB = 5). Covariance matrix uncertainties were calculated simultaneously with the recommended cross-section data by means of the PADE-2 code.

Six-digit eigenvalues for the relative covariance matrix in File-33 are as follows:

4.13473E-07	4.27626E-07	4.51208E-07	4.82385E-07
5.24529E-07	5.68750E-07	6.26389E-07	6.81740E-07
7.43528E-07	8.13479E-07	8.73417E-07	9.37226E-07
1.00863E-06	1.07570E-06	1.13454E-06	1.19653E-06
1.26664E-06	1.33965E-06	1.46617E-06	1.65554E-06
1.94736E-06	2.31852E-06	2.67098E-06	3.22483E-06
3.77462E-06	4.41698E-06	5.55287E-06	6.09540E-06
7.44285E-06	9.25147E-06	1.07387E-05	1.17943E-05
1.41203E-05	1.69054E-05	2.06437E-05	2.63415E-05
2.88626E-05	1.17262E-04	9.10324E-04	1.40176E-03
2.23894E-03	3.18553E-03	5.21607E-03	9.11440E-03
1.43199E-02	1.78563E-02	3.16668E-02	1.63229E-01
8.13509E-01			

Evaluated group cross sections and their uncertainties for the excitation function of the $^{59}Co(n,p)^{59}Fe$ reaction are given in Table 5.1. Boundaries for the neutron energy groups are the same as in File-33.

TABLE 5.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{59}Co(n,p)^{59}Fe$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 60 MeV.

Neutron energy (MeV) from – to	**Cross section (mb)**	**Uncer-tainty (%)**	**Neutron energy (MeV) from – to**	**Cross section (mb)**	**Uncer-tainty (%)**
0.796 – 3.000	0.049	90.19	15.000 – 15.500	43.280	2.25
3.000 – 3.500	0.792	7.32	15.500 – 16.000	40.917	2.89
3.500 – 4.000	2.153	5.85	16.000 – 16.500	38.913	3.51
4.000 – 4.500	4.415	5.12	16.500 – 17.000	37.221	4.01
4.500 – 5.000	7.329	4.67	17.000 – 17.500	35.781	4.39
5.000 – 5.500	10.468	4.45	17.500 – 18.000	34.534	4.70
5.500 – 6.000	13.546	4.42	18.000 – 18.500	33.429	4.99
6.000 – 6.500	16.499	4.45	18.500 – 19.000	32.429	5.32
6.500 – 7.000	19.385	4.41	19.000 – 19.500	31.505	5.70
7.000 – 7.500	22.297	4.25	19.500 – 20.000	30.634	6.13
7.500 – 8.000	25.325	4.02	20.000 – 22.500	28.250	7.45
8.000 – 8.500	28.538	3.80	22.500 – 25.000	24.651	9.45
8.500 – 9.000	31.987	3.64	25.000 – 27.500	21.501	10.59
9.000 – 9.500	35.693	3.59	27.500 – 30.000	18.809	11.13
9.500 – 10.000	39.636	3.61	30.000 – 32.500	16.552	11.39
10.000 – 10.500	43.721	3.64	32.500 – 35.000	14.673	11.55
10.500 – 11.000	47.750	3.64	35.000 – 37.500	13.109	11.70
11.000 – 11.500	51.390	3.60	37.500 – 40.000	11.801	11.88
11.500 – 12.000	54.195	3.50	40.000 – 42.500	10.700	12.08
12.000 – 12.500	55.724	3.28	42.500 – 45.000	9.765	12.30
12.500 – 13.000	55.729	2.93	45.000 – 47.500	8.965	12.54
13.000 – 13.500	54.306	2.52	47.500 – 50.000	8.275	12.78
13.500 – 14.000	51.868	2.13	50.000 – 55.000	7.413	13.13
14.000 – 14.500	48.947	1.83	55.000 – 60.000	6.484	13.59
14.500 – 15.000	45.993	1.83			

Low uncertainties of 1.83% to 2.25% in the evaluated cross sections are observed in the neutron energy range from 13.5 to 15.5 MeV, while a very large uncertainty of 90.2 % from threshold to 3.00 MeV is caused by significant discrepancies in the experimental data. Over neutron energies from 4.50 to 13.5 MeV and 15.5 to 17.5 MeV, uncertainties in the cross sections vary between 2.52% and 4.67%, but these values increase from 4.70% to 13.59% over the neutron energy range from 17.5 to 60 MeV due to inadequate experimental data and uncertainties in the cross-sections predictions of the theoretical model calculations.

Fig. 5.1 compares the re-evaluated excitation function for the $^{59}Co(n,p)^{59}Fe$ reaction over the neutron energy range from threshold to 60.0 MeV with the equivalent cross sections in ENDF/B-VII.0 and the experimental data adopted in the evaluation. Data from the MENDL-2 library and the excitation functions from the new evaluations and ENDF/B-VII.0 are shown in Fig. 5.2, along with the rejected experimental data. The ENDF/B-VII.0 data were evaluated before the publication of the new experimental data of Mannhart and Schmidt [5.24] - ENDF/B-VII.0 underestimates the cross sections of the $^{59}Co(n,p)^{59}Fe$ reaction in the energy range from 6.5 to 13.5 MeV. Below neutron energies of 6.5 MeV and between 13.5 and 20.0 MeV, both evaluations are in good agreement. However, the excitation function of the $^{59}Co(n,p)^{59}Fe$ reaction from the MENDL-2 library differs significantly from the newly evaluated data, especially for incident neutron energies above 15 MeV, and also disagrees with the ENDF/B-VII.0 evaluation at these higher energies. Experimental data of Kim *et al.* over the energy range of 45 to 75 MeV overestimate the cross section values significantly [5.20] in comparison with equivalent cross sections predicted from theoretical model calculations - a multiplication factor of Fc = 0.13048 is required for these experimental data to be brought into reasonable agreement with the results of the new evaluation (see Fig. 5.2).

Integral experimental data for the $^{59}Co(n,p)^{59}Fe$ reaction are given in Refs. [5.35-5.43]. Seven experiments were carried out in neutron fields similar to the ^{235}U thermal fission neutron spectrum [5.35-5.41], while two experiments were performed in a ^{252}Cf spontaneous fission neutron spectrum [5.42, 5.43].

Measured integral cross sections for the ^{235}U thermal fission neutron spectrum range from 0.35 to 2.10 mb [5.35-5.41]. The lowest value of 0.35 mb was obtained by Rochlin [5.35], and no information on the uncertainty is given in this publication. A value of (2.1 ± 0.3) mb was measured by Bushuev *et al.* [5.40] - this value was recalculated to the fission spectrum from data measured in the core of the BN-350 reactor operating with highly enriched fuel. The results of Refs. [5.37, 5.39, 5.41] agree within their experimental uncertainties. An average cross section of 1.46 mb determined by Braun and Nagy [5.37] was measured with a large uncertainty of 23%, and therefore the more accurate values for $\langle\sigma\rangle_{U\text{-}235}$ of (1.396 ± 0.033) mb as measured by Mannhart [5.39] and (1.382 ± 0.053) mb as obtained by Horibe *et al.* [5.41] were effectively emphasised in the present study. Horibe *et al.* used an enriched ^{235}U fission plate converter in their measurements.

The experimental data obtained in a ^{252}Cf spontaneous fission neutron spectrum by Dezsö and Csikai of (1.916 ± 0.083) mb [5.42] and by Mannhart of (1.690 ± 0.040) mb [5.43] differ by 14.4%. However, the average cross section determined by Dezsö and Csikai was not taken into account in the benchmark calculations because such a large value can not be suitably derived from representative microscopic experimental data.

Mannhart analyzed the experimental data for both spectra and recommended a values of (1.396 ± 0.033) mb for the ^{235}U thermal fission neutron spectrum, and (1.690 ± 0.042) mb for the ^{252}Cf

spontaneous fission neutron spectrum [5.44]. These evaluated experimental data were used in benchmark calculations.

The results of tests with the re-evaluated excitation function for the $^{59}Co(n,p)^{59}Fe$ reaction are given in Table 5.2. Calculated average cross sections from this work are compared with the equivalent ENDF/B-VII.0 and MENDL-2 data. C/E values show that the integral cross sections calculated from the re-evaluated excitation function and ENDF/B-VII.0 microscopic data agree well with the experimental data for both benchmark neutron spectra, while the equivalent data calculated from MENDL-2 are significantly discrepant.

TABLE 5.2. CALCULATED AND MEASURED AVERAGE CROSS SECTIONS FOR THE $^{59}Co(n,p)^{59}Fe$ REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	Average cross section, mb		C/E
	Calculated	Measured	
^{235}U thermal fission neutron spectrum	1.4194 [A]	1.396 ± 0.033 [6.49]	1.01676
	1.4125 [B]		1.01182
	0.90698 [C]		0.64970
^{252}Cf spontaneous fission neutron spectrum	1.7155 [A]	1.690 ± 0.042 [6.49]	1.01509
	1.6926 [B]		1.00154
	1.1347 [D]		0.67142

[A] present evaluation.
[B] ENDF/B-VII.0
[C] MENDL-2.

Discrepancies between the calculated and measured integral cross sections are within the limits of the experimental uncertainties and the ENDF/B-VII.0 evaluation. The results of the new evaluation in the energy range from 6.5 to 13.5 MeV agree well with the microscopic experimental data of Mannhart and Schmidt [5.24], Huang Xialong *et al.* [5.19] and corrected experimental data of Smith and Meadows [5.3]; however, the ENDF/B-VII.0 evaluated data over this neutron energy range are noticeable lower than these particular experimental data. C/E values reflect the slight overestimation of the evaluated integral experimental data for both spectra.

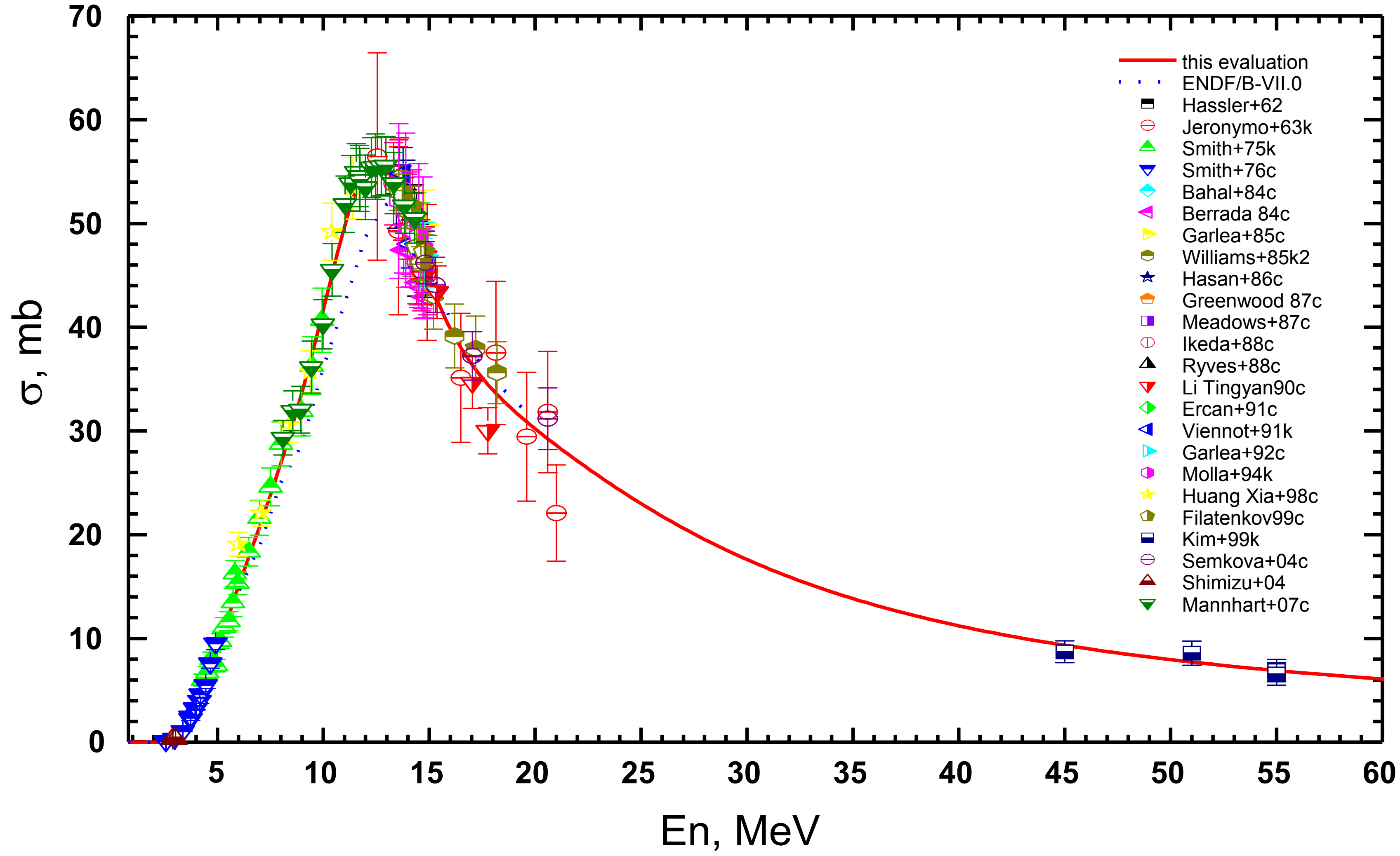

FIG. 5.1. Re-evaluated excitation function of the $^{59}Co(n,p)^{59}Fe$ reaction in the energy range from threshold to 60 MeV in comparison with ENDF/B-VII.0 and experimental data.

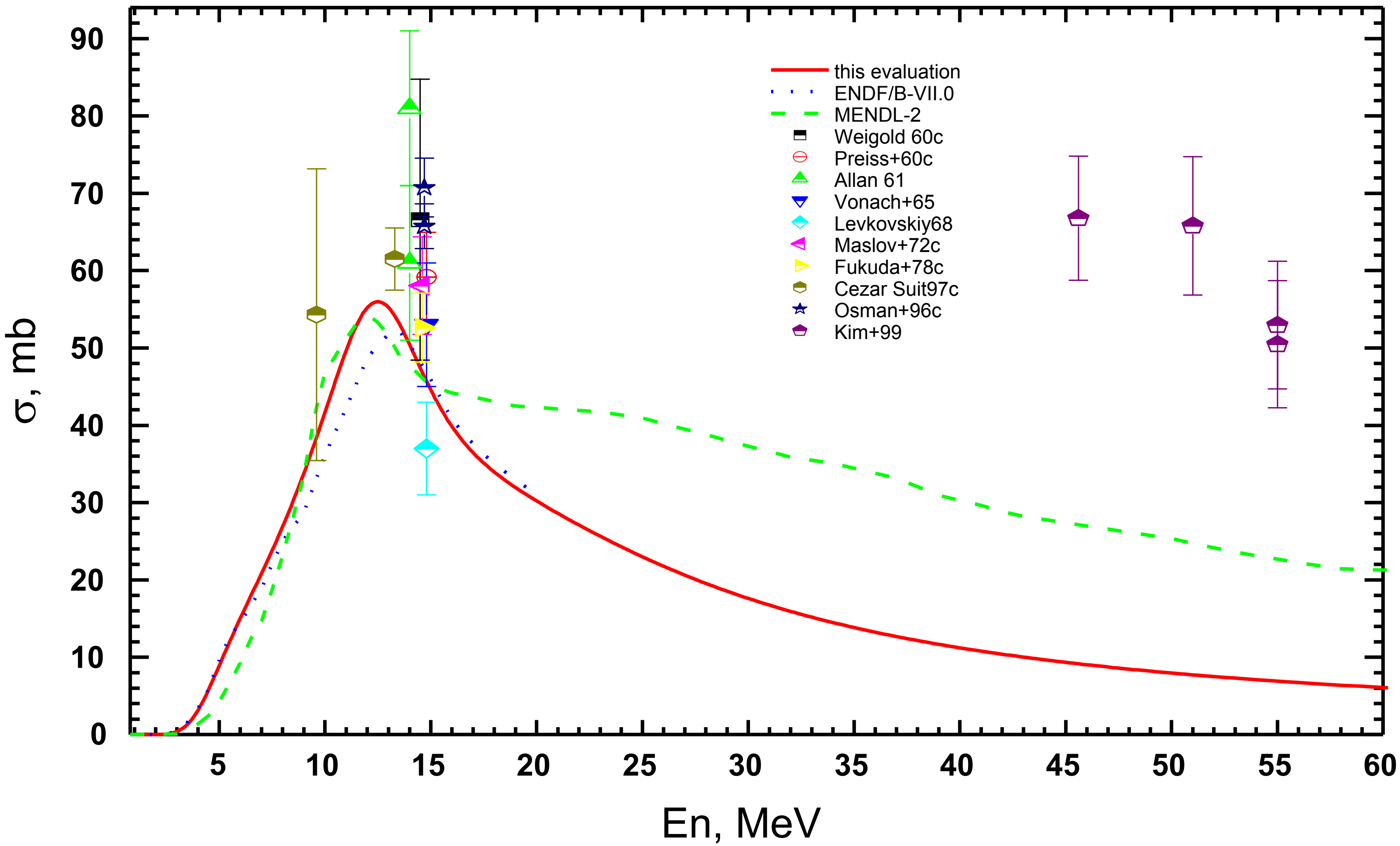

FIG. 5.2. Re-evaluated excitation function of the $^{59}Co(n,p)^{59}Fe$ reaction in the energy range from threshold to 60 MeV in comparison with ENDF/B-VII.0, MENDL-2 and rejected experimental data.

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6. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE $^{59}Co(n,2n)^{58m+g}Co$ REACTION

The isotopic abundance of ^{59}Co in natural cobalt is 100 atom percent. A 24.889-keV (J_{π} = 5+) metastable level of ^{58}Co is populated in the (n,2n) reaction, and undergoes 100% IT decay with a half-life of (9.04 ± 0.11) hours, and the emission of a 24.889-keV gamma transition (I_{γ} = 0.000389 ± 0.000012) and 0.78, 6.915 ($K_{\alpha2}$), 6.930 ($K_{\alpha1}$), 7.649 ($K_{\beta3}$) and 7.649-keV ($K_{\beta1}$) X-ray radiation (most intense X-ray emissions are $K_{\alpha2}$ (I_x = 0.080 ± 0.004) and $K_{\alpha1}$ (I_x = 0.158 ± 0.009)). Ground state ^{58g}Co undergoes 100% EC decay with a half-life of (70.86 ± 0.06) days; 511-keV annihilation radiation (I_{γ} = 0.298 ± 0.004) and 810.759-keV gamma radiation (I_{γ} = 0.99450 ± 0.00010) can be used to determine the $^{59}Co(n,2n)^{58m+g}Co$ reaction rate. Reaction rate measurements should be performed after the metastable state has undergone significant decay (T_m ≥ 90.4 hours after the end of irradiation). Recommended decay data for the half-lives, and X-ray and gamma-ray emission probabilities per decay of ^{58m}Co and ^{58g}Co were taken from Ref. [2.6] of Section 2.

Microscopic experimental data were analyzed during the preparation of the input database assembled in order to evaluate the cross sections and uncertainties of the $^{59}Co(n,2n)^{58m+g}Co$ reaction [6.1-6.45]. During this procedure, experimental data given in Refs. [6.2-6.6, 6.8, 6.9, 6.11, 6.14-6.25, 6.27, 6.29, 6.31-6.33, 6.39, 6.41-6.44] were corrected in terms of the newly recommended cross-section data for the monitor reactions used in the measurements and the recommended decay data (see Table 2.1). Careful analysis of the experimental cross-section data for the $^{59}Co(n,2n)^{58m+g}Co$ reaction between 11 and 14 MeV shows that the most precisely measured data were obtained by Frehaut *et al.* [6.11] and Mannhart and Schmidt [6.33]. The measurements of Kern *et al.* [6.1] were renormalized to a value of 409.2 mb at 12.62 MeV as determined from these experimental data. Data of Cezar Suita *et al.* [6.29] were also corrected by a factor of Fc = 1.14499 by taking the more precise data into account.

Analysis of the experimental cross-section data for the $^{59}Co(n,2n)^{58m+g}Co$ reaction between 13 and 15 MeV indicates that the most reliable data were measured by Weigold [6.2], Wenush and Vonach [6.4], Weigold and Glover [6.5], Barrall *et al*. [6.8], Fukuda *et al.* [6.10], Hasan *et al.* [6.17], Meadows *et al.* [6.18], Ikeda *et al.* [6.20], Kimura and Kobayashi [6.22], Li Tingyan *et al.* [6.23], Garlea *et al.* [6.25], Iwasaki *et al*. [6.26], and Molla *et al.* [6.27]. More recent experimental data of Semkova *et al*. [6.32] and Mannhart and Schmidt [6.33] agree with these data within their uncertainties. Cross sections obtained by Ikeda et al. [6.20] in two sets of measurements were used as the reference data for correction of the experimental data from Refs. [6.7, 6.14]. The original data of Okumura [6.7] and Huang Jianzhou *et al.* [6.14] were corrected with respect to the new standards data, and renormalized to the integral cross section of Ikeda *et al.* [6.20] in the overlapping energy range from 13.39 to 14.94 MeV; correction factors for these experimental data were Fc = 0.71961 and 0.95521, respectively.

Above a neutron energy of 15 MeV, the most representative experimental data are those of Semkova *et al*. [6.32]. Data of Paulsen and Liskien [6.6] were corrected with respect to the new standards and multiplied by a factor of Fc = 1.08842, as determined from two ratios:

3. ratio of the cross-section integrals of Ikeda *et al.* [6.20] and Paulsen and Liskien [6.6] in the energy range of 13.34 to 14.94 MeV, R = 1.088206, and
4. ratio of the cross-section integrals of Semkova *et al.* [6.32] and Paulsen and Liskien [6.6] in the energy range of 14.81 to 19.36 MeV, R = 1.088633.

Cross-section data given in Refs. [6.34-6.45] were rejected due to their large deviations from a significant amount of the other extensive experimental data – rejected experimental data in Refs. [6.34-6.36, 6.38, 6.40-6.41, 6.43, 6.45] had only been measured at one or two neutron energy points from 14 to 15 MeV.

The excitation function for the $^{59}Co(n,2n)^{58m+g}Co$ reaction in the energy region from threshold to 60 MeV was evaluated by means of statistical analyses of the experimental cross-section data [6.1-6.33] and theoretical modelling calculations. Uncertainties in the evaluated excitation function for the $^{59}Co(n,2n)^{58m+g}Co$ reaction are given in the form of a relative covariance matrix for 48-neutron energy groups (LB = 5). Covariance matrix uncertainties were calculated simultaneously with the recommended cross-section data by means of the PADE-2 code.

Six-digit eigenvalues for the relative covariance matrix in File-33 are as follows:

1.17519E-05	1.18901E-05	1.20954E-05	1.23231E-05
1.25599E-05	1.28152E-05	1.31251E-05	1.35366E-05
1.41638E-05	1.53005E-05	1.70968E-05	1.92698E-05
2.37801E-05	2.92459E-05	3.75758E-05	4.54525E-05
5.84343E-05	7.15999E-05	8.22985E-05	9.41809E-05
1.08914E-04	1.24863E-04	1.41512E-04	1.58718E-04
1.76290E-04	1.93113E-04	2.04246E-04	2.17762E-04
2.36636E-04	2.56933E-04	2.77951E-04	2.99381E-04
3.31092E-04	3.69081E-04	4.06286E-04	4.54022E-04
5.01935E-04	5.11280E-04	5.73614E-04	6.49448E-04
7.36019E-04	8.31088E-04	1.42913E-03	2.23904E-03
7.86541E-03	2.03901E-02	9.83103E-02	9.97381E-01

All of these eigenvalues are positive.

Evaluated group cross sections and their uncertainties for the $^{59}Co(n,2n)^{58m+g}Co$ reaction are listed in Table 6.1. Group boundaries are the same as in File-33. These data show that the smallest uncertainties in the evaluated cross sections of 0.91% to 0.97% are observed over the neutron energy range from 13.5 to 15.5 MeV. Evaluated cross sections in the energy intervals from 12.0 to 13.5 MeV and 15.5 to 19.0 MeV may also be defined as well-determined. A significant uncertainty of 15.25% in the cross sections from threshold to 11.5 MeV arises from the large uncertainties in and discrepancies between the experimental data within this region. Experimental cross-section data for neutron energies above 21 MeV are only reported in Refs. [6.28, 6.30] with significant uncertainties. Theoretical modelling calculations are unable to provide satisfactory cross sections for the $^{59}Co(n,2n)^{58m+g}Co$ reaction better than 15% to 50% accuracy for incident neutron energies of 20 to 60 MeV because of the inadequacies in the input data. Thus, the uncertainty in the evaluated excitation function increases from 2.3% at 20 MeV to 26.35% between 57.5 and 60.0 MeV neutron energy.

Figs 6.1 compares the re-evaluated excitation function for the $^{58}Co(n,2n)^{58m+g}Co$ reaction over the neutron energy range from threshold to 60.0 MeV with the equivalent cross sections of IRDF-2002, ENDF/B-VII.0, MENDL-2 and experimental data. Evaluated cross sections and rejected experimental data are shown in Fig. 6.2. The IRDF-2002 and newly-evaluated data agree resonably well over the energy range from threshold to 20 MeV, while the ENDF/B-VII.0 evaluation underestimates the cross sections in the energy range from 15 to 20 MeV. The MENDL-2 evaluation also underestimates the cross sections from 17 to 21 MeV and above 25 MeV neutron energy.

Integral experiments for the $^{58}Co(n,2n)^{58m+g}Co$ reaction are described in Refs. [6.46-6.52]. Five experiments were carried out in neutron fields with similar spectra to the ^{235}U thermal fission neutron spectrum [6.46-6.50], and two experiments were performed in a ^{252}Cf spontaneous fission neutron spectrum [6.51, 6.52]. Experimental data obtained for ^{235}U thermal fission neutron spectrum and ^{252}Cf spontaneous fission neutron spectrum were corrected with respect to the newly recommended cross sections for the monitor reactions and decay data.

TABLE 6.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{59}Co(n,2n)^{59m+g}Co$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 60 MeV.

Neutron energy (MeV) from to	**Cross section (mb)**	**Uncer-tainty (%)**	**Neutron energy (MeV) from to**	**Cross section (mb)**	**Uncer-tainty (%)**
10.633 - 11.500	50.177	15.25	25.000 - 26.000	390.335	13.02
11.500 - 12.000	198.320	2.09	26.000 - 27.000	350.999	14.24
12.000 - 12.500	329.264	1.68	27.000 - 28.000	320.665	15.25
12.500 - 13.000	454.548	1.36	28.000 - 29.000	296.384	16.12
13.000 - 13.500	563.193	1.11	29.000 – 30.000	276.331	16.91
13.500 - 14.000	650.035	0.96	30.000 - 31.000	259.352	17.64
14.000 - 14.500	714.646	0.91	31.000 - 32.000	244.690	18.31
14.500 - 15.000	759.749	0.91	32.000 - 33.000	231.834	18.93
15.000 - 15.500	789.555	0.97	33.000 - 34.000	220.423	19.50
15.500 - 16.000	808.535	1.11	34.000 - 35.000	210.194	20.03
16.000 - 16.500	820.731	1.29	35.000 - 36.000	200.951	20.53
16.500 - 17.000	829.449	1.46	36.000 - 37.000	192.542	20.99
17.000 - 17.500	837.138	1.60	37.000 - 38.000	184.849	21.41
17.500 - 18.000	845.261	1.72	38.000 - 39.000	177.776	21.81
18.000 - 18.500	854.072	1.85	39.000 - 40.000	171.246	22.18
18.500 - 19.000	862.236	2.01	40.000 – 42.000	162.383	22.69
19.000 - 19.500	866.513	2.12	42.000 – 44.000	151.872	23.31
19.500 - 20.000	862.022	2.19	44.000 - 46.000	142.664	23.85
20.000 - 20.500	843.746	2.35	46.000 - 48.000	134.523	24.33
20.500 - 21.000	809.167	2.89	48.000 - 50.000	127.270	24.77
21.000 - 22.000	731.471	4.22	50.000 - 52.500	120.013	25.20
22.000 - 23.000	615.727	6.69	52.500 - 55.000	112.852	25.63
23.000 - 24.000	516.716	9.29	55.000 - 57.500	106.500	26.01
24.000 - 25.000	443.366	11.43	57.500 – 60.000	100.827	26.35

Measured integral cross sections for the ^{235}U thermal fission neutron spectrum range from 0.185 to 0.340 mb [6.46-6.50]. The lowest value of (0.185 ± 0.015) mb was obtained by Sekine and Baba [6.48], while the higher value of (0.340 ± 0.030) mb was determined by Nasyrov and Sciborskij [6.46]. Results from three studies agree within the limits of their experimental uncertainties [6.47, 6.48, 6.49], and with the average cross section $<\sigma>_{U\text{-}235}$ of (0.2028 ± 0.0800) mb evaluated by Mannhart [6.53].

Experimental data obtained in a ^{252}Cf spontaneous fission neutron spectrum by Dezsö and Csikai [6.51] of (0.554 ± 0.030) mb and by Mannhart [6.52] of (0.408 ± 0.010) mb differ by 6%. The average cross section determined by Dezsö and Csikai was not taken into account in the benchmark calculations because such a large value can not be suitably derived from representative microscopic experimental data. Mannhart recommended an average cross section for the ^{252}Cf spontaneous fission neutron spectrum $<\sigma>_{Cf\text{-}252}$ of (0.4051 ± 0.0073) mb [6.53]. Evaluated experimental data from Ref. [6.53] for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were used in the benchmark calculations. Tests were made on the data as given in Table 6.2, where C/E is the ratio of the calculated to experimental cross sections. C/E values show that the smallest discrepancies between the calculated and experimental data are obtained for the newly evaluated and ENDF/B-VII.0 data (approximately -1.3% for ^{235}U thermal fission neutron spectrum and +0.7% for ^{252}Cf spontaneous fission neutron spectrum). The IRDF-2002 evaluations are shown to be inferior to the newly evaluated and ENDF/B-VII.0 data. Average cross sections calculated from the MENDL-2 excitation function are seriously discrepant with respect to the experimental data for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra (8.6% and 5.8%, respectively).

TABLE 6.2. CALCULATED AND MEASURED AVERAGE CROSS SECTIONS FOR THE ^{59}Co(n,2n)$^{58m+g}$Co REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	Average cross section, mb		C/E
	Calculated	Measured	
^{235}U thermal fission neutron spectrum	0.19999 [A]	0.2028 ± 0.0800 [6.53]	0.98614
	0.20829 [B]		1.02707
	0.20029 [C]		0.98762
	0.18542 [D]		0.91430
^{252}Cf spontaneous fission neutron spectrum	0.40799 [A]	0.4051 ± 0.0102 [6.53]	1.00713
	0.42292 [B]		1.04399
	0.40780 [C]		1.00667
	0.38171 [D]		0.94226

[A] present evaluation.
[B] IRDF-2002 (IRDF-90 version 2).
[C] ENDF/B-VII.0.
[D] MENDL-2.

Evaluated data from the ENDF/B-VII.0 library have approximately the same C/E value as the present evaluation, but do not agree with the experimental data of Semkova *et al.* [6.32]. As a result of this significant discrepancy, the ENDF/B-VII.0 data underestimate systematically the cross section of the ^{58}Co(n,2n)$^{58m+g}$Co reaction at neutron energies between 15 and 20 MeV.

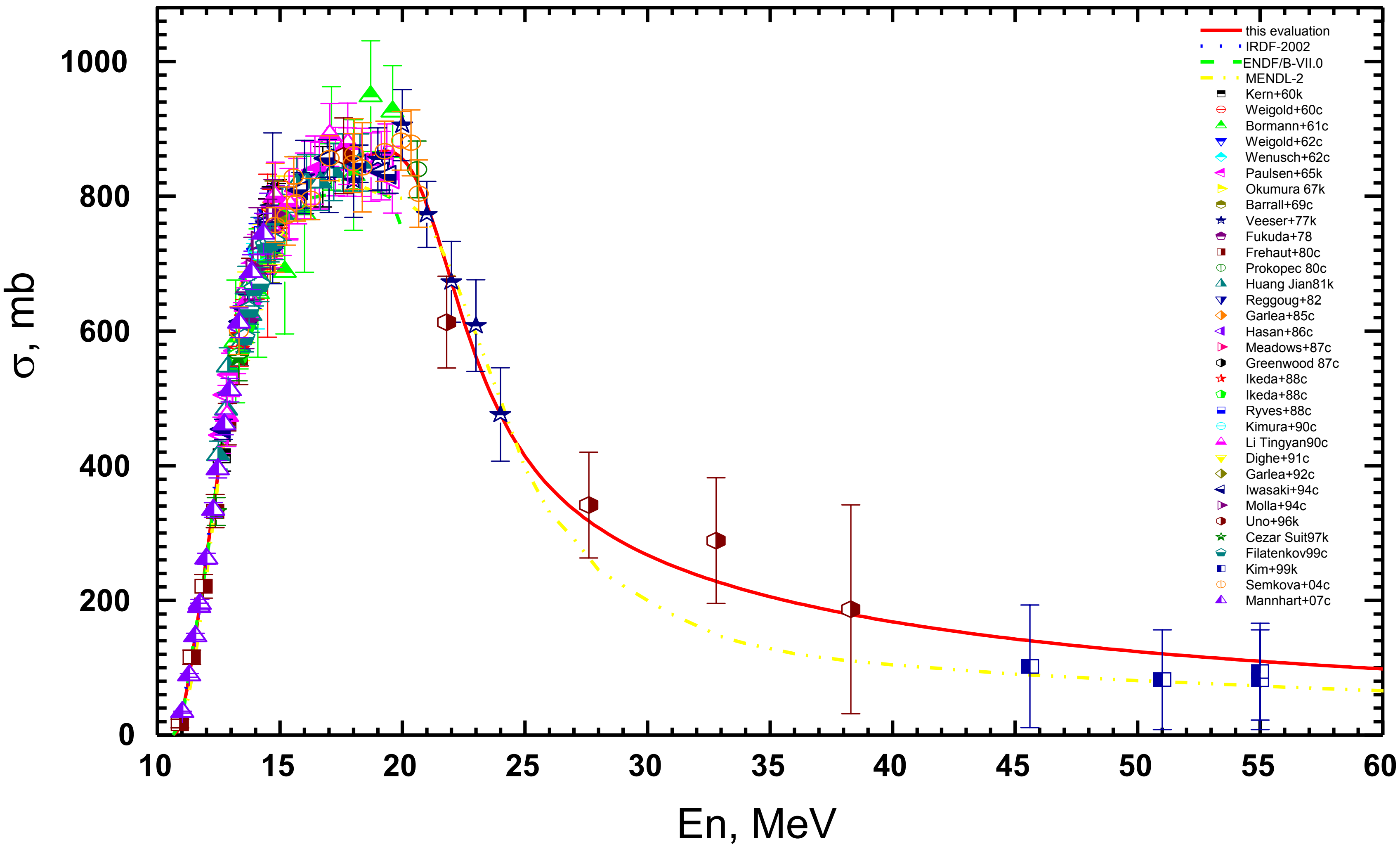

FIG. 6.1. Re-evaluated excitation function of the $^{59}Co(n,2n)^{58m+g}Co$ reaction in the energy range from threshold to 60 MeV in comparison with IRDF-2002, ENDF/B-VII.0, MENDL-2 and experimental data.

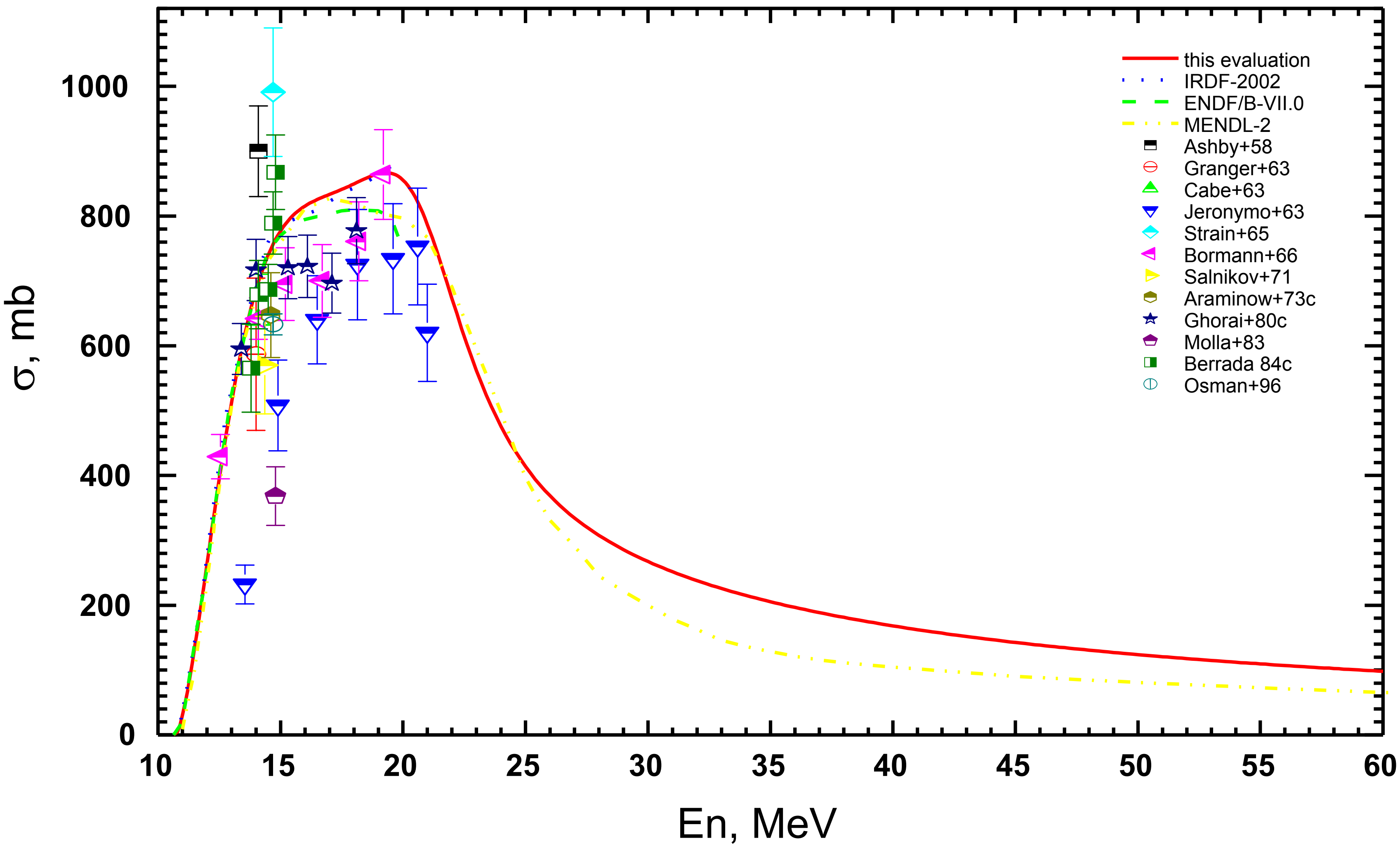

FIG. 6.2. Re-evaluated excitation function of the $^{59}Co(n,2n)^{58m+g}Co$ reaction in the energy range from threshold to 60 MeV in comparison with IRDF-2002, ENDF/B-VII.0, MENDL-2 and rejected experimental data.

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7. RE-EVALUATION OF THE EXCITATION FUNCTION OF THE ^{90}Zr(n,2n)$^{89m+g}$Zr REACTION

The abundance of the ^{90}Zr isotope in natural zirconium is 51.45 ± 0.40 atom percent. A 587.8-keV (J_{π}=1/2-) metastable level of ^{89}Zr is populated in the (n,2n) reaction, and undergoes (93.77 ± 0.12)% IT decay and 6.23% EC decay with a half-life of (4.161 ± 0.017) minutes. Both the 587.8-keV (I_{γ} = 0.8964 ± 0.0012) and 1507.4-keV gamma radiation (I_{γ} = 0.0606 ± 0.0018) can be used to determine the ^{90}Zr(n,2n)^{89m}Zr reaction rate. Ground state ^{89g}Zr undergoes 100% EC decay with a half-life of (78.41 ± 0.12) hours; 511-keV annihilation radiation (I_{γ} = 0.455 ± 0.005) and 909.15-keV gamma radiation (I_{γ} = 0.9904 ± 0.0001) are frequently used to determine the ^{90}Zr(n,2n)$^{89m+g}$Zr reaction rate. Recommended decay data for the half-lives and gamma-ray emission probabilities per decay of ^{89m}Zr and ^{89g}Zr were taken from Ref. [2.6] of Section 2.

Microscopic experimental data were analyzed in the preparation of the input database for the evaluation of the cross sections and uncertainties of the ^{90}Zr(n,2n)$^{89m+g}$Zr reaction [7.1-7.43]. During this procedure, the experimental data of Refs. [7.1-7.5, 7.8-7.14, 7.16-7.21, 7.23] were corrected with respect to the newly recommended cross-section standards and decay data (see Table 2.1).

Careful analysis of the experimental cross-section data for the ^{90}Zr(n,2n)$^{89m+g}$Zr reaction between 13 and 15 MeV indicates that the most reliable data in this energy range have been measured by Pavlik *et al.* [7.9], Iguchi *et al.* [7.15], Ikeda *et al.* [7.22] and Filatenkov and Chuvaev [7.23]. Csikai [7.10] and Ikeda *et al.* [7.11] measured relative ^{27}Al(n,α)^{24}Na reaction cross sections [7.4], and the experimental data of Refs. [7.2, 7.5, 7.10, 7.12, 7.13, 7.14, 7.17, 7.19, 7.21] are in good agreement with these data after correction with respect to the new standards.

The results of the precise measurements of Pavlik *et al.* in the neutron energy range 13.435 to 14.830 MeV [7.9] and the equivalent data of Filatenkov and Chuvaev from 13.56 to 14.78 MeV [7.23] were used as reference data in the correction of experimental data from Refs. [7.1, 7.6]. Experimental data of Nethaway obtained in the neutron energy range 13.67 to 14.81 MeV [7.6] were renormalized by a factor Fc = 0.94580. Prestwood and Bayhurst measured over a wide energy region of 12.13 to 19.76 MeV [7.1], and these data were renormalized as described below.

An additional correction was applied to the data reported in Refs. [7.1, 7.3, 7.8]. The original experimental data of Prestwood and Bayhurst [7.1], Rieder and Muenzer [7.3] and Bayhurst *et al.* [7.8] were not corrected for the 6.32% decay of the metastable state to ^{89}Y - their measured cross sections are the sum of ($\sigma_g + 0.9377 \cdot \sigma_m$), where σ_g and σ_m are the cross sections of the ^{90}Zr(n,2n)^{89g}Zr and ^{90}Zr(n,2n)^{89m}Zr reactions, respectively. The last component of this cross section was determined from the excitation function of the ^{90}Zr(n,2n)^{89m}Zr reaction evaluated in Ref. [7.44].

Cross-section data from Refs. [7.25-7.43] were rejected due to their large discrepancies when compared with all other experimental data; furthermore, cross-section data from Refs. [7.25-7.26, 7.30, 7.32-7.35, 7.38-7.39, 7.41, 7.43] were only measured at one energy point from 14 to 15 MeV.

The excitation function for the ^{90}Zr(n,2n)$^{89m+g}$Zr reaction in the neutron energy range from threshold to 40 MeV was evaluated by means of a comprehensive statistical analysis of the experimental cross-section data [7.1-7.24] and theoretical modelling calculations. Uncertainties in the evaluated excitation function are given in the form of a relative covariance matrix for 46-neutron energy groups (LB = 5). Covariance matrix uncertainties were calculated simultaneously with the recommended cross-section data by means of the PADE-2 code.

Six-digit eigenvalues for the relative covariance matrix of File-33 are as follows:

6.23724E-07	6.27795E-07	6.33912E-07	6.44490E-07
6.59408E-07	6.75080E-07	6.98174E-07	7.30611E-07
7.60534E-07	8.03916E-07	8.71141E-07	9.20493E-07
9.81385E-07	1.03561E-06	1.10034E-06	1.16750E-06
1.22867E-06	1.32391E-06	1.45806E-06	1.60547E-06
1.71646E-06	1.94412E-06	2.27292E-06	2.47336E-06
2.94345E-06	3.56946E-06	4.01120E-06	5.32616E-06
5.91221E-06	8.04621E-06	9.54973E-06	1.16616E-05
1.56257E-05	1.90592E-05	2.13849E-05	5.41284E-05
2.94052E-04	4.57017E-04	7.37718E-04	3.06321E-03
3.25375E-03	6.52383E-03	8.70755E-03	2.59362E-02
9.20734E-02	4.86308E-01		

Evaluated group cross sections and their uncertainties for the $^{90}Zr(n,2n)^{89m+g}Zr$ reaction are listed in Table 7.1. Group boundaries are the same as in File-33. While the lowest uncertainties in the evaluated cross sections of 0.81% to 1.00% are observed in the neutron energy range from 13.6 to 15.6 MeV, a significant uncertainty of 7.88% occurs from threshold to 12.4 MeV due to the large uncertainties and discrepancies between experimental data in this region. Experimental cross-section data for the neutron energies above 20 MeV are only reported in Ref. [7.8], with significant uncertainties. Theoretical modelling calculations are unable to provide satisfactory cross sections for the $^{90}Zr(n,2n)^{89m+g}Zr$ reaction better than 10% to 40% accuracy for incident neutron energies between 20 and 40 MeV. Thus, the uncertainty in the evaluated excitation function increases from 2.3% at 20 MeV to 34.88% between 38 and 40 MeV. However, the evaluated excitation function in the energy range from 13.5 to 15.5 MeV is recommended as a reference for cross-section data in activation measurements.

The re-evaluated excitation function for the $^{90}Zr(n,2n)^{89m+g}Zr$ reaction in the neutron energy range from 10.0 to 40.0 MeV is compared in Fig. 7.1 with the equivalent data from IRDF-2002, ENDF/B-VII.0, MENDL-2, GNASH and experimental data. Fig. 7.2 shows the excitation functions and all rejected experimental data, as well as the results of GNASH theoretical modelling calculations. Data from the new evaluation, IRDF-2002, ENDF/B-VII.0 and GNASH agree reasonably well up to 15 MeV, extending up to 18 MeV for the re-evaluated and ENDF/B-VII.0 data. The excitation function of the $^{90}Zr(n,2n)^{89m+g}Zr$ reaction in the MENDL-2 library contradicts all experimental data and the evaluations.

Integral experiments for the $^{90}Zr(n,2n)^{89m+g}Zr$ reaction are described in Refs. [7.45-7.53]. Most of these experiments were carried out in neutron fields with similar spectra to the ^{235}U thermal fission neutron spectrum [7.45-7.51], and only two experiments were performed in a ^{252}Cf spontaneous fission neutron spectrum [7.52, 7.53]. Experimental data obtained for ^{235}U thermal fission neutron spectrum and ^{252}Cf spontaneous fission neutron spectrum were corrected with respect to the newly recommended cross sections for the monitor reactions and decay data.

TABLE 7.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{90}Zr(n,2n)^{89m+g}Zr$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 40 MeV.

Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)	Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)
12.104 - 12.400	17.688	7.88	18.000 - 18.500	1168.722	1.70
12.400 - 12.600	65.555	3.79	18.500 - 19.000	1190.028	1.85
12.600 - 12.800	125.444	3.11	19.000 - 19.500	1206.338	2.01
12.800 - 13.000	196.230	2.48	19.500 - 20.000	1217.848	2.21
13.000 - 13.200	272.411	1.82	20.000 - 20.500	1224.590	2.47
13.200 - 13.400	349.579	1.34	20.500 - 21.000	1226.460	2.80
13.400 - 13.600	424.719	1.10	21.000 - 22.000	1219.070	3.46
13.600 - 13.800	495.979	1.00	22.000 - 23.000	1191.220	4.54
13.800 - 14.000	562.385	0.96	23.000 - 24.000	1040.820	5.59
14.000 - 14.200	623.560	0.91	24.000 - 25.000	1068.350	6.33
14.200 - 14.400	679.509	0.86	25.000 - 26.000	976.871	6.66
14.400 - 14.600	730.510	0.82	26.000 - 27.000	872.146	6.87
14.600 - 14.800	776.771	0.81	27.000 - 28.000	761.741	7.67
14.800 - 15.000	818.833	0.82	28.000 - 29.000	653.435	9.52
15.000 - 15.200	857.054	0.86	29.000 - 30.000	553.625	12.09
15.200 - 15.400	891.824	0.92	30.000 - 31.000	466.429	14.87
15.400 - 15.600	923.506	0.98	31.000 - 32.000	393.575	17.67
15.600 - 15.800	952.426	1.04	32.000 - 33.000	334.909	20.64
15.800 - 16.000	978.878	1.10	33.000 - 34.000	289.093	23.96
16.000 - 16.500	1019.587	1.19	34.000 - 35.000	254.258	27.54
16.500 - 17.000	1068.860	1.31	35.000 - 36.000	228.411	30.94
17.000 - 17.500	1109.080	1.44	36.000 - 38.000	203.137	34.23
17.500 - 18.000	1141.980	1.56	38.000 - 40.000	184.827	34.88

Measured integral cross sections for the ^{235}U thermal fission neutron spectrum range from 0.0795 to 0.245 mb [7.45-7.51]. A value of (0.0795 ± 0.0067) mb was obtained by Brodskaja *et al.* [7.46], while the highest value of (0.245 ± 0.016) mb was determined by Kobayashi *et al.* [7.45]. All other studies agree within 23% [7.47-7.51]. The integral cross section $<\sigma>_{U-235}$ as evaluated by Mannhart is (0.1027 ± 0.0028) mb [7.54], and is mainly based on his own experimental data [7.48]. Measurements of Kimura and Kobayashi carried out with 90%-enriched ^{235}U fission plate converter give an integral cross section of (0.08535 ± 0.0054) mb [7.50].

The experimental data obtained in a ^{252}Cf spontaneous fission neutron spectrum by Dezsö and Csikai of (0.255 ± 0.014) mb [7.52] and by Mannhart of (0.221 ± 0.006) mb [7.53] differ by 15.4%. Mannhart recommended an average cross section $<\sigma>_{Cf-252}$ for the ^{252}Cf spontaneous fission neutron spectrum of (0.2210 ± 0.0064) mb [7.54]. Average cross sections were calculated from four different excitation functions for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra, and were compared with the evaluated experimental data [7.54]. The resulting C/E ratios are listed in Table 7.2 - C/E values obtained for ^{235}U thermal fission neutron spectrum show significat discrepancies between calculated and experimental data.

TABLE 7.2. CALCULATED AND MEASURED AVERAGE CROSS SECTIONS FOR THE $^{90}Zr(n,2n)^{89m+g}Zr$ REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	Average cross section, mb		C/E
	Calculated	Measured	
^{235}U thermal fission neutron spectrum	0.093438 [A]	0.1027 ± 0.0028 [7.54]	0.9098
	0.095307 [B]		0.9280
	0.096500 [C]		0.9396
	0.034594 [D]		0.3369
^{252}Cf spontaneous fission neutron spectrum	0.21708 [A]	0.2210 ± 0.0064 [7.54]	0.9823
	0.22136 [B]		1.0016
	0.22321 [C]		1.0100
	0.088373 [D]		0.3999

[A] present evaluation.
[B] IRDF-2002 (IRDF-90 version 2).
[C] ENDF/B-VII.0.
[D] MENDL-2.

The average cross sections calculated from the new evaluation, IRDF-2002 and ENDF/B-VII.0 data for the ^{252}Cf spontaneous fission neutron spectrum agree well with the equivalent experimental data of Mannhart [7.54] (C/E = 0.9823-1.0100). The lowest discrepancy is observed for IRDF-2002 data (C/E = 1.0016), but the excitation function obtained in this evaluation followed the trend predicted by the well-defined experimental data of Pavilik *et al.* [7.9], while the IRDF-2002 excitation function is somewhat higher over the neutron energy range from 12.0 to 17.80 MeV. Average cross sections calculated from the MENDL-2 excitation function are discrepant with respect to the experimental data for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra (66% and 60%, respectively).

New precise measurements of the integral cross sections of the $^{90}Zr(n,2n)^{89m+g}Zr$ reaction are required for ^{235}U thermal fission neutron spectrum in order to understand the reason for the large discrepancies that exist between the calculated and experimental data.

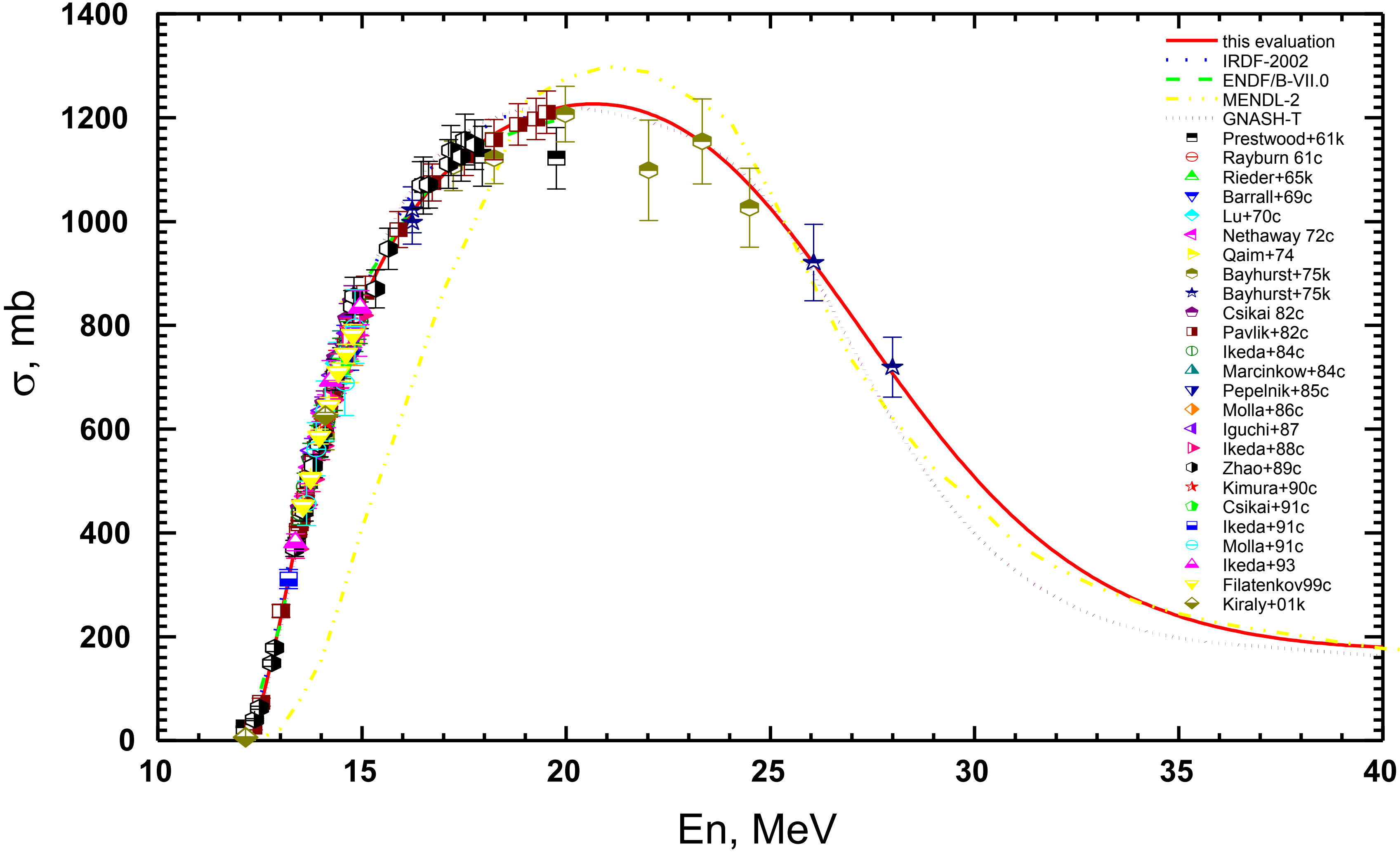

FIG. 7.1. Re-evaluated excitation function of the $^{90}Zr(n,2n)^{89m+g}Zr$ reaction in the energy range from threshold to 40 MeV in comparison with IRDF-2002, ENDF/B-VII.0, MENDL-2, GNASH and experimental data.

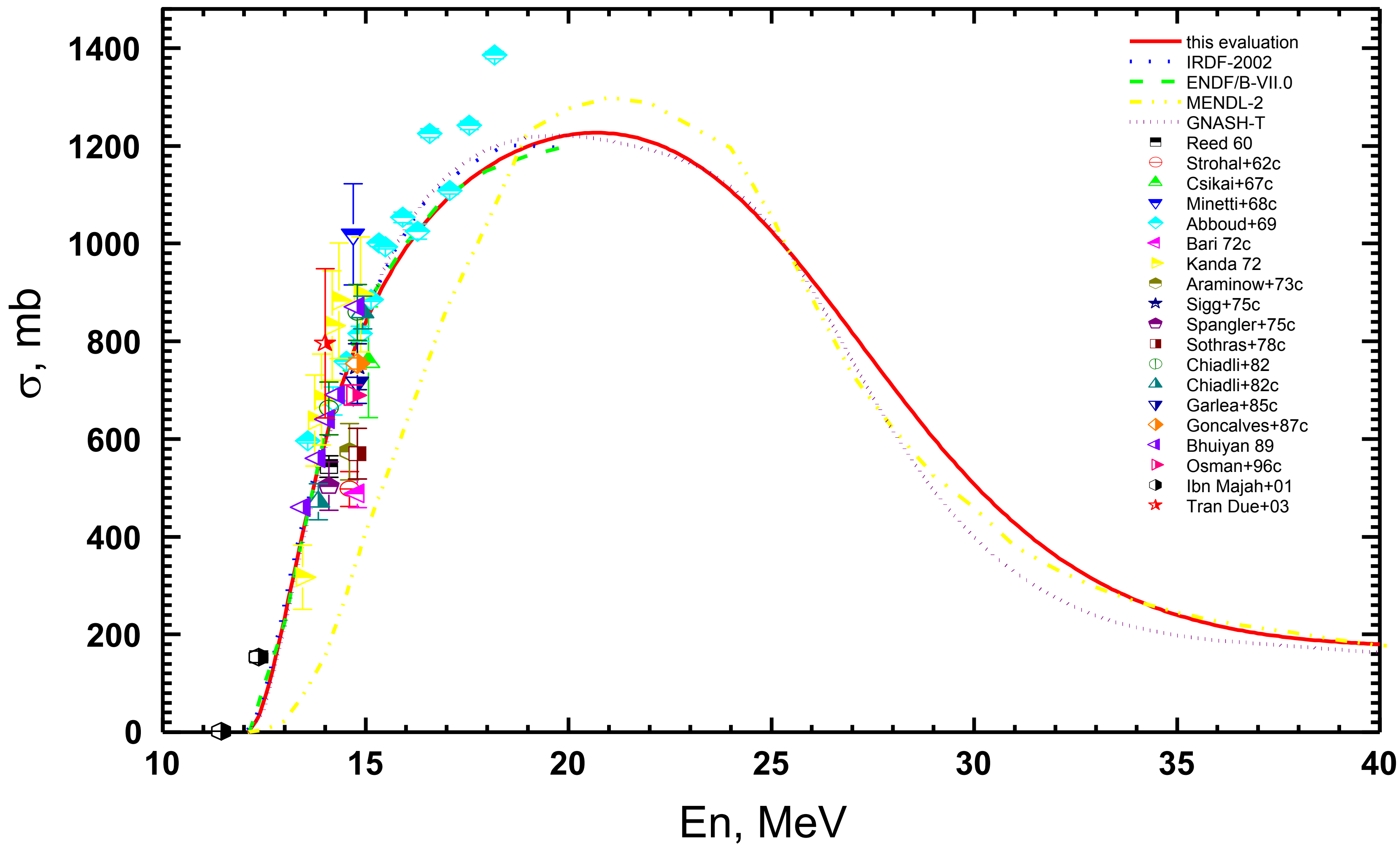

FIG. 7.2. Re-evaluated excitation function of the ^{90}Zr(n,2n)$^{89m+g}$Zr reaction in the energy range from threshold to 40 MeV in comparison with IRDF-2002, ENDF/B-VII.0, MENDL-2, GNASH and rejected experimental data.

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8. CONCLUSIONS

Re-evaluations of cross sections and their uncertainties have been carried out for five dosimetry reactions. Excitation functions for the ^{27}Al(n,α)^{24}Na, ^{55}Mn(n,2n)^{54}Mn and ^{90}Zr(n,2n)$^{89m+g}$Zr reactions were re-evaluated over the neutron energy range from threshold to 40 MeV, while the excitation functions of the ^{59}Co(n,p)^{59}Fe and ^{59}Co(n,2n)$^{58m+g}$Co reactions were re-evaluated in the energy range from threshold to 60 MeV. Compared with IRDF-2002, the upper neutron energy boundary was increased from 20 to 40 MeV for the ^{27}Al(n,α)^{24}Na, and ^{90}Zr(n,2n)$^{89m+g}$Zr reactions and from 20 to 60 MeV for the ^{59}Co(n,2n)$^{58m+g}$Co reaction. Uncertainties in the cross sections for all of these reactions are given in the form of relative covariance matrices.

Benchmark calculations performed for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra show that the integral cross sections calculated from the newly evaluated excitation functions exhibit improved agreement with related experimental data when compared with the equivalent data from the IRDF-2002, ENDF/B-VII.0, MENDL-2 libraries and the Karlsruhe-2007 evaluation. Thus, the resulting ^{27}Al(n,α)^{24}Na, ^{55}Mn(n,2n)^{54}Mn, ^{59}Co(n,p)^{59}Fe, ^{59}Co(n,p)$^{58m+g}$Co and ^{90}Zr(n,2n)$^{89m+g}$Zr cross-section files in ENDF-6 format should be considered as suitable candidates in the preparation of an improved version of the International Reactor Dosimetry File (IRDF).

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