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EVALUATION OF SOME (n,n'), (n,γ), (n,p), (n,2n) AND (n,3n) REACTION EXCITATION FUNCTIONS FOR FISSION AND FUSION REACTOR DOSIMETRY APPLICATIONS

Evaluation of the excitation functions for the $^{54}Fe(n,p)^{54}Mn$, $^{58}Ni(n,2n)^{57}Ni$, $^{67}Zn(n,p)^{67}Cu$, $^{92}Mo(n,p)^{92m}Nb$, $^{93}Nb(n,\gamma)^{94}Nb$, $^{113}In(n,n')^{113m}In$, $^{115}In(n,\gamma)^{116m}In$, and $^{169}Tm(n,3n)^{167}Tm$ reactions

Progress Report on Research Contract No 16242

K.I. Zolotarev, P.K. Zolotarev

Institute of Physics and Power Engineering,
Obninsk, Russia

December 2013

IAEA Nuclear Data Section
Vienna International Centre, P.O. Box 100, 1400 Vienna, Austria

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International Atomic Energy Agency
Vienna International Centre
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1. INTRODUCTION

Cross section data for the ^{54}Fe(n,p)^{54}Mn, ^{58}Ni(n,2n)^{57}Ni, ^{67}Zn(n,p)^{67}Cu, ^{92}Mo(n,p)^{92m}Nb, ^{93}Nb(n,γ)^{94}Nb, ^{113}In(n,n')^{113m}In, ^{115}In(n,γ)^{116m}In, ^{169}Tm(n,3n)^{167}Tm reactions are needed to solve a wide spectrum of scientific and technical tasks. Activation detectors based on these reactions may be used in the field of reactor dosimetry. Furthermore, the ^{54}Fe(n,p)^{54}Mn reaction is often used in experimental nuclear physics as a monitor reaction for measurements of unknown cross sections by means of the activation method over the neutron energy range from 5 to 15 MeV. The ^{93}Nb(n,γ)^{94}Nb reaction is also very promising for using in retrospective neutron dosimetry for determination of total neutron fluence during a campaign of a reactor.

In the existing version of the International Reactor Dosimetry File [1.1] and the new extended version named as IRDFF [1.2] data for excitation functions of ^{67}Zn(n,p)^{67}Cu, ^{92}Mo(n,p)^{92m}Nb, ^{113}In(n,n')^{113m}In, and ^{169}Tm(n,3n)^{167}Tm reactions are absent. Data for these reactions are also absent in the JENDL/D-99 dosimetry file [1.3]. Excitation functions of ^{67}Zn(n,p)^{67}Cu and ^{169}Tm(n,3n)^{167}Tm are presented in the TENDL-2012 [1.4], EAF-2010 [1.5], JENDL-4.0 [1.6], JEFF-3.1/A [1.7], MENDL-2 [1.8] libraries. Cross section data for the ^{67}Zn(n,p)^{67}Cu reaction up to 20 MeV are given also in the JENDL/HE-2007 library [1.9]. Excitation functions of the ^{92}Mo(n,p)^{92m}Nb and ^{113}In(n,n')^{113m}In reactions are evaluated in the EAF-2010 and JEFF-3.1/A libraries. Cross section data for the ^{113}In(n,n')^{113m}In reaction are given also in the TENDL-2010 library [1.10]. It is necessary to note that neutron data in the JEFF-3.1/A and JENDL-4.0 libraries were evaluated up to 20 MeV. Neutron data in the TENDL-2012, EAF-2010, MENDL-2 and TENDL-2010 libraries had been evaluated up to 30 MeV, 60 MeV, 100 MeV and 200 MeV, respectively. Neutron cross sections in the MENDL-2, TENDL-2010 and TENDL-2012 libraries had been obtained on the basis of pure theoretical model calculations and are not appropriate for reactor and fusion dosimetry application.

The main aims of this work were the evaluation of the cross section data and related uncertainty covariance matrixes for ^{54}Fe(n,p)^{54}Mn, ^{58}Ni(n,2n)^{57}Ni, ^{67}Zn(n,p)^{67}Cu, ^{92}Mo(n,p)^{92m}Nb, ^{93}Nb(n,γ)^{94}Nb, ^{113}In(n,n')^{113m}In, ^{115}In(n,γ)^{116m}In, and ^{169}Tm(n,3n)^{167}Tm reactions. Neutron data for the ^{54}Fe(n,p)^{54}Mn, ^{58}Ni(n,2n)^{57}Ni, ^{67}Zn(n,p)^{67}Cu, ^{92}Mo(n,p)^{92m}Nb and ^{169}Tm(n,3n)^{167}Tm reactions were evaluated with extension to higher neutron energies up to 30 – 60 MeV. New evaluations were performed on the basis of correction to the new standards of all available experimental data and data obtained from consistent theoretical model calculations.

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2. METHOD OF EVALUATION OF EXCITATION FUNCTIONS FOR DOSIMETRY REACTIONS

2.1. Analysis of experimental data

Differential and integral experimental data for the ^{54}Fe(n,p)^{54}Mn, ^{58}Ni(n,2n)^{57}Ni, ^{67}Zn(n,p)^{67}Cu, ^{92}Mo(n,p)^{92m}Nb, ^{93}Nb(n,γ)^{94}Nb, ^{113}In(n,n')^{113m}In, ^{115}In(n,γ)^{116m}In and ^{169}Tm(n,3n)^{167}Tm dosimetry reactions were taken mainly from the EXFOR library. Any information absent in the EXFOR Data file was taken from the original publications.

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. The standards used to correct the microscopic experimental data under investigation are given below 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	Carlson+ [2.1]			
^{6}Li(n,t)^{4}He	Carlson+ [2.1]			
^{27}Al(n,α)^{24}Na	Zolotarev [2.2]	14.997 (12) h	Gamma 1368.63 keV	0.999936(15) [2.11]
^{27}Al(n,p)^{27}Mg	Zolotarev+ [2.3]	9.458 (12) m	Gamma 843.76 keV Gamma 1014.44 keV	0.718 (4) [2.12] 0.280 (4) [2.12]
^{56}Fe(n,p)^{56}Mn	Zolotarev+ [2.5]	2.5789 (1) h	Gamma 846.754 keV Gamma 1810.72 keV	0.989 (3) [2.12] 0.272 (8) [2.12]
^{58}Ni(n,p)^{58}Co	Zolotarev+ [2.6]	70.86 (6) d	Gamma 511 keV Gamma 810.759 keV	0.298 (4) [2.12] 0.99450 (10) [2.12]
^{63}Cu(n,2n)^{62}Cu	Zolotarev [2.7]	9.673 (8) m	Beta+ 2925.8 keV Gamma 511 keV Gamma 1173.02 keV	0.9720 (2) [2.13] 1.9486 (5) [2.13] 0.00342 (5) [2.13]
^{65}Cu(n,2n)^{64}Cu	Zolotarev [2.7]	12.700 (2) h	Beta+ 653.1 keV Beta– 578.7 keV Gamma 511 keV Gamma 1345.77 keV	0.1740 (22) [2.14] 0.390 (3) [2.14] 0.348 (4) [2.14] 0.00473 (10) [2.14]
^{64}Zn(n,p)^{64}Cu	Zolotarev [2.7]	12.700 (2) h	Beta+ 653.1 keV Beta– 578.7 keV Gamma 511 keV Gamma 1345.77 keV	0.1740 (22) [2.14] 0.390 (3) [2.14] 0.348 (4) [2.14] 0.00473 (10) [2.14]
^{93}Nb(n,2n)^{92m}Nb	Zolotarev [2.8]	10.15 (2) d	Gamma 934.44 keV	0.9907 (4) [2.12]
^{75}As(n,2n)^{74}As	IRDF-2002 [2.4]	17.77 (2) d	Gamma 511 keV Gamma 595.83 keV Gamma 634.78 keV	0.58 (6) [2.15] 0.59 (3) [2.15] 0.154 (10) [2.15]
^{90}Zr(n,2n)^{89}Zr	Zolotarev [2.2]		Gamma 511 keV Gamma 909.15 keV	0.455 (5) [2.16] 0.9904 (1) [2.16]
^{115}In(n,n')^{115m}In	Zolotarev [2.9]	4.486 (4) h	Gamma 336.24 keV	0.458 (22) [2.17]
^{115}In(n,2n)^{114m}In	Zolotarev [2.7]	49.51 (1) d	Gamma 190.27 keV	0.1556 (15) [2.18]
^{127}I(n,γ)^{128}I	Zolotarev [2.10]	24.99 (2) m	Gamma 442.90 keV	0.169 (17) [2.12]
^{169}Tm(n,2n)^{168}Tm	Zolotarev [2.8]	93.1 (2) d	Gamma 184.30 keV Gamma 198.25 keV Gamma 447.52 keV Gamma 815.99 keV	0.1745 (56) [2.12] 0.5240 (160) [2.12] 0.2306 (71) [2.12] 0.4899 (150) [2.12]
^{197}Au(n,2n)^{196}Au	Zolotarev [2.7]	6.183 (10) d	Gamma 333.03 keV Gamma 355.73 keV Gamma 426.10 keV	0.229 (6) [2.12] 0.870 (4) [2.12] 0.066 (4) [2.12]
^{197}Au(n,γ)^{198}Au	Carlson+ [2.1]	2.6947 (3) d	Gamma 411.802 keV	0.9562 (4) [2.19]
^{235}U(n,f)	Carlson+ [2.1]			
^{238}U(n,γ)	Carlson+ [2.1]	23.45 (2) m	Gamma 74.664 keV	0.492 (4) [2.20]
^{238}U(n,f)	Carlson+ [2.1]			

In Table 2.1 for Beta transitions the $E_{\beta max}$ values are listed.
Thermal cross sections and resonance integrals for (n,γ) and (n,f) reactions evaluated by S. Mughabghab were used as recommended data [2.21].
Recommended cross section data for the monitor reactions used in measurements of integral cross sections in ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were taken from Refs. [2.22] and [2.23]. Digital data for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were taken from Refs. [2.24] and [2.25], respectively. Information about the isotopic compositions of the elements was taken from Ref. [2.26].

Corrections to the experimental data based on the new standards lead as a rule to reductions in the discrepancies and thus resulted in decreases in the uncertainties of the evaluated cross sections.

2.2. 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 and in the case of absence of experimental data. Hence, theoretical calculations were carried out to determine the excitation functions of the $^{54}Fe(n,p)^{54}Mn$, $^{58}Ni(n,2n)^{57}Ni$, $^{67}Zn(n,p)^{67}Cu$, $^{92}Mo(n,p)^{92m+g}Nb$, $^{93}Nb(n,\gamma)^{94}Nb$, $^{113}In(n,n')^{113m}In$, $^{115}In(n,\gamma)^{116m}In$ and $^{169}Tm(n,3n)^{167}Tm$ reactions from 3 - 20 MeV.

The optical-statistical method was used for a theoretical description of the excitation function of the above mention dosimetry 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 [2.27, 2.28] and EMPIRE-2.19 code [2.29]. A modified version of the GNASH code 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 coupled-channel deformed optical model code was used for these calculations [2.30], and the optical coefficients of the proton- and alpha-particle penetrabilities were determined by means of the SCAT2 code [2.31].

The data on discrete levels parameters for ^{54}Fe, ^{58}Ni, ^{67}Zn, ^{92}Mo, ^{93}Nb, ^{113}In, ^{115}In, ^{169}Tm, and all residual nuclei were obtained from Ref. [2.12]. 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. [2.32].

Continuum level densities were represented by means of the Gilbert-Cameron model [2.33] based on the Cook parameters [2.34] (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 systematic [2.35]. Recommended parameters for the giant dipole resonances were taken from Ref. [2.36].

The modified GNASH code was used to calculate the cross sections of the $^{67}Zn(n,p)^{67}Cu$, $^{92}Mo(n,p)^{92m+g}Nb$, and $^{113}In(n,n')^{113m}In$ up to 50 MeV, 40 MeV and 20 MeV, respectively. The modified GNASH code was used also to calculate the cross sections of $^{93}Nb(n,\gamma)^{94}Nb$ and $^{115}In(n,\gamma)^{116m}In$ reactions up to 20 MeV. Data for $^{54}Fe(n,p)^{54}Mn$, $^{58}Ni(n,2n)^{57}Ni$, $^{169}Tm(n,3n)^{167}Tm$ reactions were calculated by means of EMPIRE-2.19 code from threshold to 30, 60 and 50 MeV, respectively.

2.3. 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. [2.37, 2.38]. 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. [2.39]. The algorithm used to minimize $S(\vec{\beta})$ contains two approximations that simplify the calculation scheme appreciably:

1) cross section data obtained in different experiments are assumed to be uncorrelated;
2) 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 $\Delta f(E_{i_k}^k, \vec{\beta})$ are determined from the relationships:

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

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

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:

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

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_i^m (e_j^m) is the m^{th} component of the systematic uncertainty of the cross section at the i^{th} (j^{th}) point; P_{ij}^m 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 [2.37].

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3. EVALUATION OF EXCITATION FUNCTION OF THE ^{54}Fe(n,p)^{54}Mn REACTION

The isotopic abundance of ^{54}Fe in natural iron is (5.845 ± 0.035) atom percent, and the ^{54}Mn obtained via the (n,p) reaction undergoes 100% via ε capture decay mode with a half-life of (312.05 ± 0.04) days. The 834.848-keV gamma radiation (I_γ = 0.999760 ± 0.000010) is normally used to determine the ^{54}Fe(n,p)^{54}Mn reaction rate. Recommended decay data for the half-life and gamma ray emission probabilities per decay of ^{54}Mn were taken from Ref. [3.42].

Microscopic experimental data about the ^{54}Fe(n,p)^{54}Mn reaction excitation function is given in the works [3.1-3.41] and covered neutron energies from 1.71 MeV to 17.3 MeV.

Experimental data of works [3.1], [3.5-3.7], [3.10-3.11], [3.13-3.19], [3.21-3.29], [3.31-34], [3.36-3.37], [3.39-3.41] in the process of analysis were corrected to the new standards for the relevant monitor reactions (see Table 2.1) and to the recommended decay data for ^{54}Mn [3.42].

Special correction was applied to the experimental data [3.7], [3.8] and [3.16].
Corrected to the new standards experimental data of Lauber and Malmskog [3.7] were renormalized to the integral of cross section calculated from experimental data by Smith and Meadows [3.16] in the overlapping energy range 2.31-3.81 MeV.
Original experimental data by Salisbury et al. [3.8] in the energy range 2.23-16.75 MeV were corrected to the preliminary evaluated cross section of 505.0 mb at 6.18 MeV.
Correction factors for the experimental data [3.7] and [3.8] were equal to Fc = 1.41721 and Fc = 0.8938, respectively.
Data of Smith and Meadows [3.16] measured using neutrons from D(d,n)^{3}He reaction were renormalized to the results of this experiment obtained with ^{7}Li(p,n)^{7}Be neutron source in the overlapping interval 5.407 - 5.438 MeV. The D(d,n)^{3}He data in the energy range 5.407 - 9.948 MeV were multiplied to the factor Fc = 1.06614.

Data of March and Morton for total ^{54}Fe(n,p)^{54}Mn cross section at 13.5 MeV [3.2] were obtained by summation of measured compound and direct processes cross sections.
Experimental data of Carroll and Smith [3.9] were taken only for the incident neutron energies 3.55 - 6.02 MeV. Data measured at 15.7 MeV and 17.4 MeV points were not taken into account due to their discrepancy with the main bulk of experimental data.

Excitation function for the ^{54}Fe(n,p)^{54}Mn reaction in the energy region from threshold to 30 MeV was evaluated by means of statistical analysis of experimental cross section data [3.1-3.2], [3.6-3.14], [3.16-3.20], [3.22-3.26], [3.28-3.30], [3.32], [3.34], [3.36-3.41], and data from theoretical model calculation, which were obtained by means of modified GNASH code [2.27].

Evaluation of the ^{54}Fe(n,p)^{54}Mn reaction cross section and related uncertainties were performed by means of PADE-2 code. Uncertainties in the evaluated excitation function for the ^{54}Fe(n,p)^{54}Mn reaction are given in the form of a relative covariance matrix for 45-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:

6.90511E-07	6.93234E-07	6.96921E-07	7.01168E-07
7.05292E-07	7.10306E-07	7.17869E-07	7.35083E-07
7.55545E-07	7.98141E-07	8.39606E-07	9.29082E-07
1.02500E-06	1.15143E-06	1.39468E-06	1.64886E-06
1.91803E-06	2.45934E-06	3.22329E-06	4.05437E-06
4.72864E-06	5.96803E-06	7.70244E-06	9.82298E-06
1.22946E-05	1.50778E-05	1.81318E-05	2.17691E-05
2.71222E-05	3.32873E-05	3.87824E-05	4.33509E-05
4.69942E-05	5.10461E-05	2.72934E-04	1.99654E-03
2.93121E-03	3.94152E-03	4.88506E-03	5.77641E-03
9.45929E-03	1.16711E-02	2.32610E-02	3.84256E-02
4.04256E-01			

Eigenvalues of the relative covariance matrix given in File-33 were calculated by means of PADE-2 and tested in additional by COVEIG code [3.73]. It was found that all of the eigenvalues are positive. Differences in the eigenvalues determined by PADE-2 code and COVEIG code only exist for the lowest eigenvalues and do not exceeded 0.68%. The highest of the eigenvalues are equivalent.

Evaluated group cross sections and their uncertainties for the excitation function of the $^{54}Fe(n,p)^{54}Mn$ reaction are listed in Table 3.1. Group boundaries are the same as in File-33

TABLE 3.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{54}Fe(n,p)^{54}Mn$ REACTION IN THE NEUTRON ENERGY RANGE FROM 0.07 TO 30 MeV.

Neutron energy (MeV) from to	**Cross section (mb)**	**Uncer-tainty (%)**	**Neutron energy (MeV) from to**	**Cross section (mb)**	**Uncer-tainty (%)**
0.070 - 2.000	2.195	19.30	13.000 - 13.500	403.320	2.44
2.000 - 2.500	37.908	8.28	13.500 - 14.000	365.973	2.12
2.500 - 3.000	94.712	5.84	14.000 - 14.500	327.338	2.00
3.000 - 3.500	159.981	5.41	14.500 - 15.000	289.641	2.10
3.500 - 4.000	236.871	4.96	15.000 - 15.500	254.571	2.44
4.000 - 4.500	318.071	4.44	15.500 - 16.000	223.148	3.06
4.500 - 5.000	389.957	4.05	16.000 - 16.500	195.785	4.02
5.000 - 5.500	442.998	3.85	16.500 - 17.000	172.446	5.29
5.500 - 6.000	476.099	3.70	17.000 - 17.500	152.826	6.80
6.000 - 6.500	493.869	3.58	17.500 - 18.000	136.490	8.45
6.500 - 7.000	502.164	3.51	18.000 - 18.500	122.966	10.15
7.000 - 7.500	505.644	3.47	18.500 - 19.000	111.804	11.84
7.500 - 8.000	507.265	3.44	19.000 - 19.500	102.602	13.46
8.000 - 8.500	508.587	3.38	19.500 - 20.000	95.011	14.95
8.500 - 9.000	510.175	3.29	20.000 - 21.000	86.156	16.86
9.000 - 9.500	511.885	3.21	21.000 - 22.000	77.458	18.91
9.500 - 10.000	513.039	3.19	22.000 - 23.000	71.424	20.34
10.000 - 10.500	512.522	3.25	23.000 - 24.000	67.191	21.29
10.500 - 11.000	508.903	3.35	24.000 - 25.000	64.192	21.86
11.000 - 11.500	500.615	3.42	25.000 - 26.000	62.049	22.18
11.500 - 12.000	486.268	3.38	26.000 - 28.000	59.951	22.36
12.000 - 12.500	465.034	3.18	28.000 - 30.000	58.299	22.39
12.500 - 13.000	437.005	2.83			

Uncertainties in the evaluated $^{54}Fe(n,p)^{54}Mn$ excitation function range from 2.00% to 22.39%. The smallest uncertainties in the evaluated cross sections 2.00-2.83% are observed in the neutron energy range from 12.5 to 15.5 MeV. Uncertainties in cross sections exceeding 10% are in the energy ranges 0.07-2.00 and 18 - 30 MeV.

The evaluated excitation function for the $^{54}Fe(n,p)^{54}Mn$ reaction in the neutron energy range from 1 MeV to 30 MeV is shown in Fig. 3.1 and Fig. 3.2 in comparison with the equivalent data from new international dosimetry file IRDFF-2012 (ENDF/B-VII.1) and corrected experimental data. The $^{54}Fe(n,p)^{54}Mn$ reaction cross sections in the JENDL/D-99 (JENDL-4.0) were evaluated only up to 20 MeV and due to this reason are not shown. As is seen in Fig 3.1 and Fig. 3.2 the present evaluation in comparison with IRDFF-2012 (ENDF/B-VII.1) evaluation agrees better with the representative experimental data. In the IRDFF-2012 (ENDF/B-VII.1) library cross section values are systematically higher in the energy ranges 2.7 - 3.0 MeV and 16 - 30 MeV. At the same time the $^{54}Fe(n,p)^{54}Mn$ reaction cross sections presented in the IRDFF-2012 (ENDF/B-VII.1) library in the energy range between 4 – 12 MeV are systematically lower in comparison with data from new evaluation.

Integral experiments for the $^{54}Fe(n,p)^{54}Mn$ reaction are described in the works [3.43-3.69]. Most of the experiments were carried out in the neutron fields with spectra which are similar to ^{235}U thermal fission neutron spectrum [3.43-3.53], [3.57], [3.60], [3.61], [3.63-3.68]. Integral cross sections measured in the field of ^{252}Cf spontaneous fission neutron spectrum presented in the works [3.54-3.60], [3.62]. Only one experiment was performed in the SIGMA-SIGMA spectrum [3.69]. Experimental data obtained for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were corrected to the new recommended cross sections for monitor reactions and decay data (see Ref. [2.22] and [2.23] in section 2).

Measured integral cross sections for ^{235}U thermal fission neutron spectrum having a range of obtained results from (58.39 ± 8.17) mb [3.49] to (94.30 ± 9.55) mb [3.46]. Data presented in Refs. [3.43-3.44], [3.46], [3.49-3.50], [3.52-3.53], [3.61], [3.64] and [3.68] were obtained in reactor cores measurements. The $^{54}Fe(n,p)^{54}Mn$ integral cross section measured in facilities with 90%-enriched ^{235}U fission plate converter (FPC) presented in works [3.45], [3.47-3.48], [3.51], [3.60], [3.63], [3.65-3.67]. In the publications [3.66] and [3.67] are given the identical experimental data.

The results of reactor core measurements were not taken into account in calculation of average-weighted cross section value for the $^{54}Fe(n,p)^{54}Mn$ reaction. Average-weighted value obtained from FPC experimental data (76.01 ± 4.03) mb [3.45], (75.01 ± 3.68) mb [3.47], (77.14 ± 2.28) mb [3.51], (79.30 ± 3.01) mb [3.60], (79.91 ± 2.29) mb [3.63], (81.20 ± 5.17) mb [3.65] and (76.02 ± 4.07) mb [3.67] is equal to $\langle\sigma\rangle_{U-235}$ = (78.09 ± 1.17) mb. Recommended in Ref. [2.22] integral cross section is equal to $\langle\sigma\rangle_{U-235}$ = (79.67 ± 1.10) mb. This value is oriented mainly at experimental data of Mannhart [3.63]. It is necessary to note also that evaluated in this work value $\langle\sigma\rangle_{U-235}$ = (78.09 ± 1.17) mb agrese well with a value $\langle\sigma\rangle_{U-235}$ = (78.50 ± 1.88) mb determined by Grigor'ev et al. from measurements at 8 different reactors [3.61].

As it was mentioned above experimental data for the $^{54}Fe(n,p)^{54}Mn$ reaction in the field of ^{252}Cf spontaneous fission neutron spectrum are presented in eight works. In the publications [3.58] and [3.62] are given the identical experimental data. Average-weighted value obtained from experimental data (87.00 ± 3.00) mb [3.54], (87.36 ± 3.84) mb [3.55], (84.60 ± 1.92) mb [3.56], (87.88 ± 5.46) mb [3.57], (88.52 ± 4.16) mb [3.58], (84.99 ± 4.33) mb [3.59] and (88.99 ± 2.67) mb [3.60] is equal to $\langle\sigma\rangle_{Cf-252}$ = (86.62 ± 1.17) mb. Recommended in Ref. [2.23] integral cross section is equal to $\langle\sigma\rangle_{Cf-252}$ = (86.84 ± 1.16) mb.

Evaluated excitation function for the ^{54}Fe(n,p)^{54}Mn reaction was tested against the above mentioned integral experimental data. Calculated averaged cross sections over ^{235}U thermal fission neutron spectrum, ^{252}Cf spontaneous fission neutron spectrum and SIGMA-SIGMA spectrum are compared with the RRDFF-2012 (ENDF/B-VII.1), JENDL/D-99 (JENDL-4.0) and experimental data in Table 3.2. Given in the last column value of C/E is the ratio of the calculated to experimental cross sections.

TABLE 3.2. CALCULATED AND MEASURED AVERAGED CROSS SECTIONS FOR THE ^{54}Fe(n,p)^{54}Mn REACTION IN THE THREE NEUTRON SPECTRA.

Type of neutron field	**Averaged cross section, mb**		**C/E**
	Calculated	**Measured**	
^{235}U thermal fission neutron spectrum	78.257 [A]	78.09 ± 1.17 [*]	1.00214
	80.180 [B]		1.02676
	80.946 [C]		1.03657
^{252}Cf spontaneous fission neutron spectrum	86.578 [A]	86.62 ± 1.17 [*]	0.99952
	88.151 [B]		1.01767
	89.241 [C]		1.03026
SIGMA-SIGMA neutron spectrum	16.867 [A]	18.20 ± 0.90 [3.69]	0.92676
	17.465 [B]		0.95962
	17.563 [C]		0.96500

[A] - Present evaluation.
[B] - RRDFF-2012 (ENDF/B-VII.1).
[C] - JENDL/D-99 (JENDL-4.0).
[*] - Average-weighted value evaluated in this work from corrected to the new standards integral experimental data.

The C/E values obtained for the ^{235}U thermal fission neutron spectrum and ^{252}Cf spontaneous fission neutron spectrum show that the ^{54}Fe(n,p)^{54}Mn integral cross sections calculated from newly evaluated excitation function are in excellent agreement with the relevant experimental data. The $\langle\sigma\rangle_{U\text{-}235}$ and $\langle\sigma\rangle_{Cf\text{-}252}$ cross sections calculated from the RRDFF-2012 (ENDF/B-VII.1) and JENDL/D-99 (JENDL-4.0) excitation functions agree worse with the relevant experimental values. Discrepancies between calculated and experimental values for these libraries are exceeding uncertainty in the experimental values. The 90%-response range for the ^{54}Fe(n,p)^{54}Mn reaction in the ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are very similar in the ranges 2.4 - 7.7 MeV and 2.40 - 8.1 MeV, respectively. The excellent agreement between calculated and measured $\langle\sigma\rangle_{U\text{-}235}$ and $\langle\sigma\rangle_{Cf\text{-}252}$ cross sections permit to confirm that the newly evaluated ^{54}Fe(n,p)^{54}Mn reaction excitation function is well determined in the energy region 2.40 - 8.1 MeV.

SIGMA-SIGMA is coupled to a thermal-fast URANIUM + BORON CARBIDE spherical source assembly located within a conventional graphite thermal column. SIGMA-SIGMA facilities were operated at three laboratories in Europe. One of the facilities has been in operation since 1970 at CEN/SCK, Mol, Belgium. The second SIGMA-SIGMA facility was operated at Institute for Nuclear Technology (ITN) Romania. The third SIGMA-SIGMA facility – NISUS was operated at ULRC, London, Great Britain. Within uncertainties of the order of 1 - 2% it has been experimentally determined, that neutron fields at three facilities are practically identical in terms of fundamental fission rates and threshold reaction rates [3.69-3.70].

The recommended neutron spectrum in SIGMA-SIGMA neutron field was determined as a result of careful investigations which had been performed at SIGMA-SIGMA (MOL) facility. These investigations were described by J. Grundl and C. Eisenhauer in Ref. [3.71]. The neutron spectrum in SIGMA-SIGMA (MOL) has been measured by three laboratories GFK (Karlsruhe), RCN (Petten), and CEN/SCK (Mol). Proton-recoil spectra were measured by all three laboratories with spherical proportional counters. In additional, ^{3}He(n,p)T and ^{6}Li(n,α)T spectrometer measurements have been performed by GFK and CEN/SCK respectively. General agreement among the observed spectra is ± 5% between 20 keV and 4 MeV. In addition to the experimental data information about neutron spectrum in a wide neutron energy range 0.4 eV – 15 MeV was obtained from calculations. The base neutron spectrum calculation was a 100-group ANISN transport calculation carried out with using ENDF/B-III cross sections. The departures between spectrometry and the base ENDF/B-III calculation were below ± 15%. The assigned spectrum for SIGMA-SIGMA [3.72] was obtained from combination of spectrometry measurements and discrete-ordinates calculations. Evaluated in the work [3.72] neutron spectrum in SIGMA-SIGMA field had been transformed in IAEA Nuclear Data Section to the SAND-II energy groups presentation and prepared as a data file in the ENDF/B format.

The integral cross section $\langle\sigma\rangle_{\Sigma\text{-}\Sigma}$ in SIGMA-SIGMA neutron field for the compared evaluations were calculated with using neutron spectrum from data file prepared in NDS, IAEA.

Data given in Table 3.2 show that calculated and measured integral cross sections $\langle\sigma\rangle_{\Sigma\text{-}\Sigma}$ in SIGMA-SIGMA neutron spectrum are significant differing, C/E= 0.92676 (new evaluation), C/E = 0.95962 (RRDFF-2012 (ENDF/B-VII.1) evaluation) and C/E= 0.96500 (JENDL/D-99 (JENDL-4.0)). The big differences (C/E = 0.92676) between calculated and experimental values $\langle\sigma\rangle_{\Sigma\text{-}\Sigma}$ can't be explained due to underestimation of the ^{54}Fe(n,p)^{54}Mn reaction cross section in the new evaluation. The 90%-response range for the ^{54}Fe(n,p)^{54}Mn reaction in SIGMA-SIGMA neutron spectrum is lying between 2.3 - 7.70 MeV. The 90%-response range for the ^{54}Fe(n,p)^{54}Mn reaction in the ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are very similar 2.4 - 7.7 MeV and 2.40 - 8.1 MeV, respectively. As was mentioned above calculated from new evaluation $\langle\sigma\rangle_{U\text{-}235}$ and $\langle\sigma\rangle_{Cf\text{-}252}$ values agree excellently with the relevant experimental data.

A rather big discrepancy between calculated and experimental values $\langle\sigma\rangle_{\Sigma\text{-}\Sigma}$ may be explained due to incorrect determination of the SIGMA-SIGMA neutron spectrum above 4 MeV. Because the SIGMA-SIGMA neutron spectrum in the energy interval 4 -15 MeV was determined from transport calculations performed using very old ENDF/B-III cross sections. It is necessary to note also that $\langle\sigma\rangle_{\Sigma\text{-}\Sigma}$ value for the ^{54}Fe(n,p)^{54}Mn reaction had been measured only in one experiment carried out by Garlea et al. at the SIGMA-SIGMA (ITN) facility [3.69].

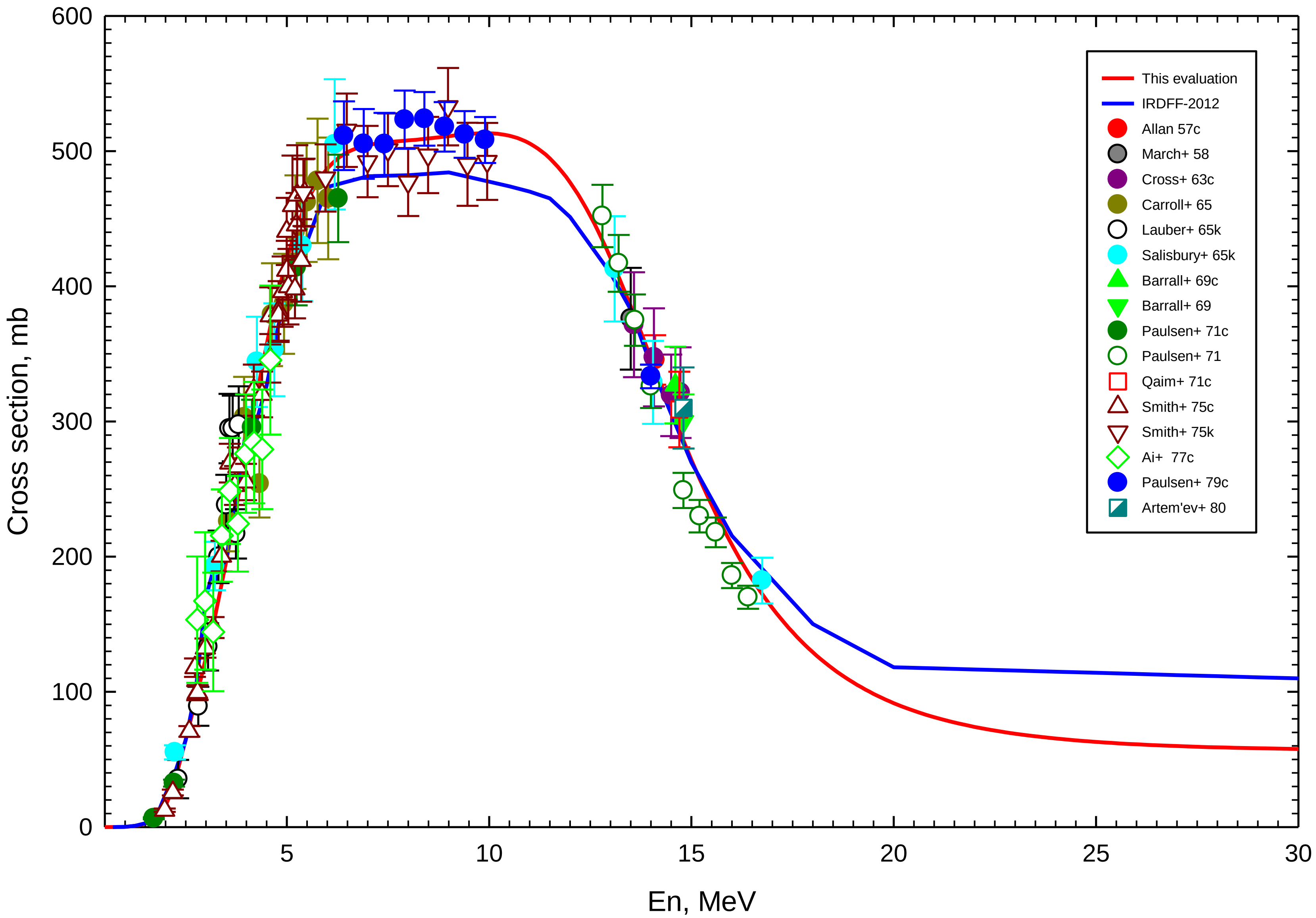

FIG. 3.1. Re-evaluated excitation function of the ^{54}Fe(n,p)^{54}Mn reaction in the energy range from threshold to 30 MeV in comparison with IRDFF-2012 and experimental data obtained between 1957 – 1980.

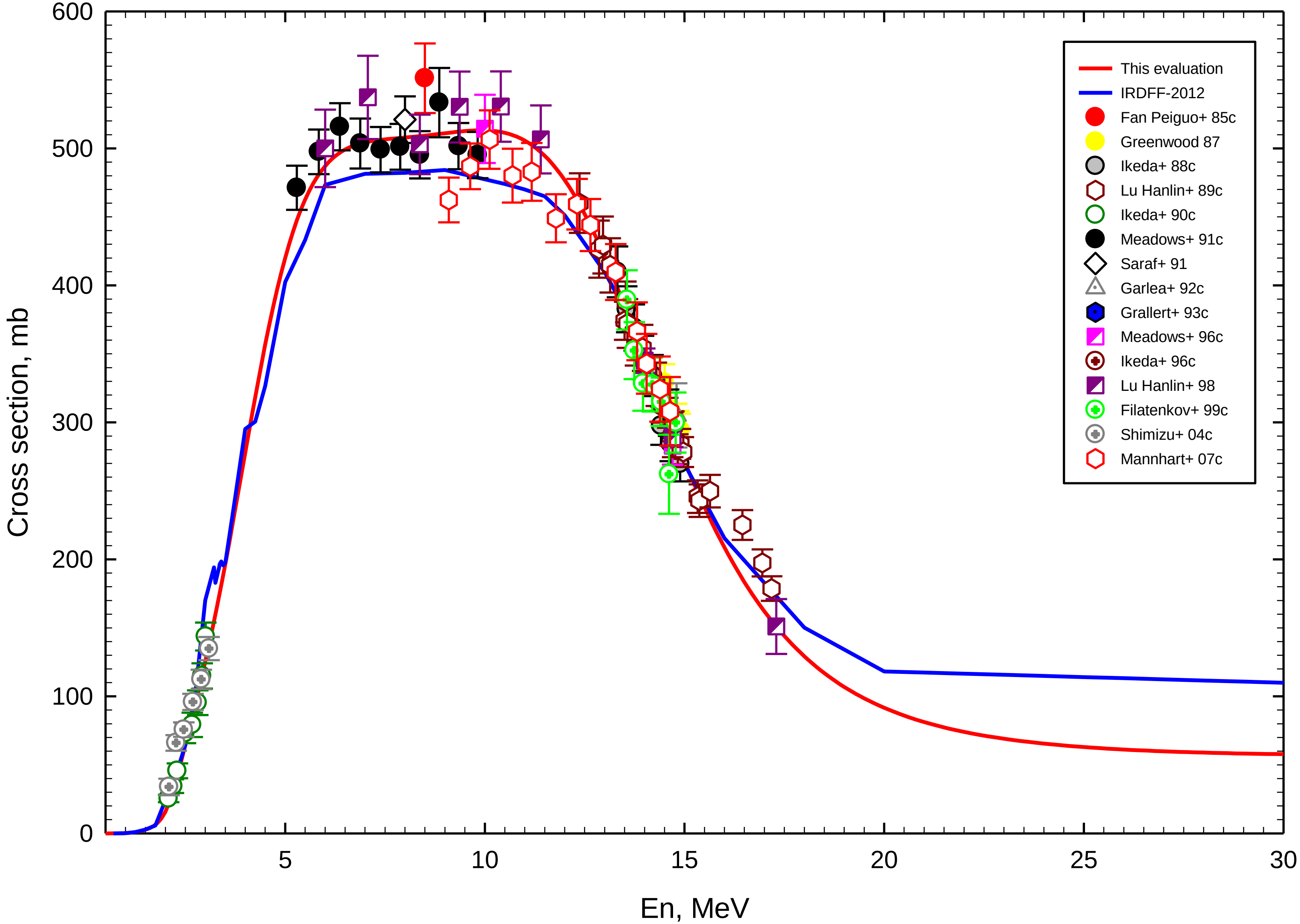

FIG. 3.2. Re-evaluated excitation function of the ^{54}Fe(n,p)^{54}Mn reaction in the energy range from threshold to 30 MeV in comparison with IRDFF-2012 and experimental data obtained between 1985 – 2007.

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4. EVALUATION OF EXCITATION FUNCTION OF THE ^{58}Ni(n,2n)^{57}Ni REACTION

The isotopic abundance of ^{58}Ni in natural nickel is (68.0769 ± 0.0089) atom percent. The residual nucleus ^{57}Ni obtained in the (n,2n) reaction undergoes 100% via β^+ decay mode with a half-life of (35.60 ± 0.06) hours. The 511-keV annihilation gamma radiation (I_γ = 0.87 ± 0.03), 1377.63-keV gamma radiation (I_γ = 0.817 ± 0.024) and 1919.52-keV gamma radiation (I_γ = 0.123 ± 0.004) are normally used to determine the ^{58}Ni(n,2n)^{57}Ni reaction rate. Recommended decay data for the half-life and gamma ray emission probabilities per decay of ^{57}Ni were taken from Ref. [4.62].

Microscopic experimental data for the ^{58}Ni(n,2n)^{57}Ni reaction excitation function is given in the works [4.1-4.61] and cover neutron energies range from 12.521 MeV to 38.3 MeV. Experimental data of works [4.3-4.8], [4.10-4.17], [4.19-4.27] and [4.29-4.61] in the process of analysis were corrected to the new standards for the relevant monitor reactions (see Table 2.1) and to the recommended decay data for ^{57}Ni.

Special correction was done with experimental data [4.21], [4.28], [4.44] and [4.55].
Original experimental data by Ngoc et al. [4.28] in the energy range 13.55-14.71 MeV were corrected to the preliminary evaluated cross section of 29.63 mb at 14.40 MeV.
After correction to the new standards experimental data by Lu Hanlin et al. [4.44] were renormalized to the preliminary evaluated cross section of 33.24 mb at 14.58 MeV.
Correction factors for the experimental data [4.28], [4.44] were equal to Fc = 1.09741 and Fc = 0.88853, respectively
Experimental data given in works [4.21] and [4.55] were corrected using representative experimental data of Pavlik et al. [4.34] and Semkova et al. [4.58].
Cross sections measured by Bayhurst et al. [4.21] were renormalized to the integral of ^{58}Ni(n,2n)^{57}Ni reaction excitation function in the energy interval 17.23 - 20.00 MeV evaluated from the experimental data [4.34] and [4.58], Fc = 0.88943.
Data of Uno et al. [4.55] measured in a wide energy range 17.60 - 38.30 MeV were corrected to the cross section value of 63.76 mb at 17.60 MeV determined from the experimental data [4.34] and [4.58], Fc = 1.12056.
Experimental data of Bormann et al. [4.13] were taken only for the incident neutron energies 12.95 - 14.90 MeV. Data measured at the 15.6 - 19.6 MeV energy points were rejected due to their discrepancy with the representative experimental data.

Excitation function for the ^{58}Ni(n,2n)^{57}Ni reaction in the energy region from threshold to 60 MeV was evaluated by means of statistical analysis of experimental cross section data [4.5-4.6], [4.7-4.8], [4.10-4.11], [4.13-4.14], [4.16-4.17], [4.19-4.21], [4.23-4.24], [4.27-4.29], [4.33-4.34], [4.36], [4.38-4.40], [4.42-4.45], [4.49-4.51], [4.53-4.58], [4.60-4.61] and data from theoretical model calculations, which were obtained by means of modified GNASH code [2.27].

Evaluation of the ^{58}Ni(n,2n)^{57}Ni reaction cross section and related uncertainties were performed by means of PADE-2 code. Uncertainties in the evaluated excitation function for the ^{58}Ni(n,2n)^{57}Ni 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. Eigenvalues of the relative covariance matrix given in File-33 were calculated by means of PADE-2 and COVEIG codes. It was found that all of the eigenvalues are positive. Differences in the eigenvalues determined by PADE-2 code and COVEIG code only exist for the lowest eigenvalues and do not exceeded 0.68%. The highest of the eigenvalues are equivalent. Six-digit eigenvalues calculated by PADE-2 code for the relative covariance matrix in File-33 are as follows:

1.10168E-06	1.14107E-06	1.19888E-06	1.26966E-06
1.34295E-06	1.42092E-06	1.51097E-06	1.62313E-06
1.72474E-06	1.85143E-06	2.00139E-06	2.12563E-06
2.30004E-06	2.50058E-06	2.68521E-06	3.08567E-06
3.48376E-06	4.18088E-06	5.00191E-06	5.70492E-06
6.77841E-06	7.35016E-06	9.00155E-06	9.77157E-06
1.16292E-05	1.35504E-05	1.45456E-05	1.70954E-05
1.98212E-05	2.13704E-05	2.31163E-05	2.60124E-05
2.90214E-05	3.20675E-05	3.50920E-05	3.64547E-05
6.84347E-05	1.51539E-04	4.28792E-04	5.22860E-04
1.20075E-03	2.60811E-03	3.67673E-03	7.30383E-03
1.41592E-02	2.46194E-02	5.31451E-02	2.07278E-01

Evaluated group cross sections and their uncertainties for the excitation function of the $^{58}Ni(n,2n)^{57}Ni$ 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 $^{58}Ni(n,2n)^{57}Ni$ REACTION IN THE NEUTRON ENERGY RANGE FROM 0.07 TO 30 MeV.

Neutron energy (MeV) from to	**Cross section (mb)**	**Uncer-tainty (%)**	**Neutron energy (MeV) from to**	**Cross section (mb)**	**Uncer-tainty (%)**
12.430 - 12.800	0.927	9.68	21.000 - 22.000	85.692	4.39
12.800 - 13.000	3.235	4.68	22.000 - 23.000	87.958	5.23
13.000 - 13.200	5.482	3.29	23.000 - 24.000	87.752	5.85
13.200 - 13.400	8.207	2.35	24.000 - 25.000	85.279	6.15
13.400 - 13.600	11.405	1.85	25.000 - 26.000	81.109	6.23
13.600 - 13.800	15.031	1.66	26.000 - 27.000	75.926	6.24
13.800 - 14.000	18.989	1.57	27.000 - 28.000	70.335	6.36
14.000 - 14.200	23.147	1.50	28.000 - 29.000	64.770	6.65
14.200 - 14.400	27.351	1.41	29.000 - 30.000	59.500	7.10
14.400 - 14.600	31.449	1.33	30.000 - 32.000	52.478	7.83
14.600 - 14.800	35.315	1.30	32.000 - 34.000	44.681	8.72
14.800 - 15.000	38.862	1.34	34.000 - 36.000	38.558	9.32
15.000 - 15.500	44.117	1.56	36.000 - 38.000	33.762	9.63
15.500 - 16.000	49.967	1.96	38.000 - 40.000	29.975	9.78
16.000 - 16.500	54.319	2.21	40.000 - 42.000	26.946	9.90
16.500 - 17.000	57.859	2.27	42.000 - 44.000	24.490	10.10
17.000 - 17.500	61.064	2.22	44.000 - 46.000	22.473	10.47
17.500 - 18.000	64.197	2.18	46.000 - 48.000	20.793	11.04
18.000 - 18.500	67.368	2.23	48.000 - 50.000	19.379	11.82
18.500 - 19.000	70.592	2.40	50.000 - 52.000	18.176	12.79
19.000 - 19.500	73.821	2.65	52.000 - 54.000	17.141	13.92
19.500 - 20.000	76.970	2.97	54.000 - 56.000	16.244	15.16
20.000 - 20.500	82.587	3.74	56.000 - 58.000	15.460	16.49
20.500 - 21.000	81.544	3.71	58.000 - 60.000	14.769	17.87

Uncertainties in the evaluated $^{58}Ni(n,2n)^{57}Ni$ excitation function range from 1.30% to 17.87%. The smallest uncertainties in the evaluated cross sections 1.30-1.96% are observed in the neutron energy range from 13.4 to 16.0 MeV. Uncertainties in the $^{58}Ni(n,2n)^{57}Ni$ reaction cross sections exceeding 10% are observed in the energy range 42 - 60 MeV.

The evaluated excitation function for the $^{58}Ni(n,2n)^{57}Ni$ reaction in the neutron energy range from threshold to 60 MeV is shown in Fig. 4.1 in comparison with the equivalent data from new international dosimetry file IRDFF-2012 (ENDF/B-VII.1) and corrected experimental data. The same information in increased scale from threshold to 15 MeV is shown in Fig. 4.2. As is seen in Fig 4.1 and Fig. 4.2 the present evaluation in comparison with IRDFF-2012 (ENDF/B-VII.1) evaluation agrees better with the representative experimental data. The $^{58}Ni(n,2n)^{57}Ni$ reaction excitation function presented in the IRDFF-2012 (ENDF/B-VII.1) library in all energy points from threshold to 60 MeV gives a systematically higher cross section value in comparison with data from new evaluation. Ratio of cross sections from IRDFF-2012 and new evaluation is equal to R= 1.30 at 30 MeV. The equivalent ratio at 60 MeV is equal to R= 3.25. Evaluated excitation functions for the $^{58}Ni(n,2n)^{57}Ni$ reaction and rejected experimental data are shown in Fig. 4.3. The $^{58}Ni(n,2n)^{57}Ni$ reaction cross sections in the JENDL/D-99 (JENDL-4.0) were evaluated only up to 20 MeV and due to this reason are not shown.

Integral experiments for the $^{58}Ni(n,2n)^{57}Ni$ reaction are described in the works [4.63-4.74]. Most experiments were carried out in the neutron fields with spectra which are similar to ^{235}U thermal fission neutron spectrum [4.63-4.67], [4.69-4.74]. Integral cross sections measured in the field of ^{252}Cf spontaneous fission neutron spectrum are presented only in one work [4.68]. Experimental data obtained for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were corrected to the new recommended cross sections for monitor reactions and decay data (see Ref. [2.22] and [2.23] in section 2).

Measured integral cross sections for ^{235}U thermal fission neutron spectrum having a range of obtained results from (3.433 ± 0.230) µb [4.66] to (5.261 ± 0.272) µb [4.64]. Data presented in Refs. [4.63-4.65], [4.67], [4.69-4.70], [4.72-4.74] were obtained in reactor core measurements. The $^{58}Ni(n,2n)^{57}Ni$ integral cross section measured in facilities with 90%-enriched ^{235}U fission plate converter (FPC) are presented in the works [4.66] and [4.71]. Data obtained in FPC measurements after applied corrections are equal to (3.433 ± 0.230) µb [4.66] and (3.817 ± 0.210) µb [4.71]. Both of the results are agreed in the limit of uncertainties. Recommended in Ref. [2.22] integral cross section is equal to $<\sigma>_{U\text{-}235}$ = (4.257 ± 0.123) µb. This value is significantly higher in comparison with FPC experimental data. It is necessary to mark also that integral cross section $<\sigma>_{U\text{-}235}$ = (3.711 ± 0.772) µb [4.63] measured by Braun and Nagy at FR-2 Karlsruhe reactor is agree well with FPC experimental data.

As it was mentioned above experimental data for the $^{58}Ni(n,2n)^{57}Ni$ reaction in the field of ^{252}Cf spontaneous fission neutron spectrum is presented only in a one work [4.68]. Data of work [4.68] after correction to the new standards gives for the $^{58}Ni(n,2n)^{57}Ni$ reaction a value of $<\sigma>_{Cf\text{-}252}$ = (8.558 ± 0.310) µb. Recommended in Ref. [2.23] integral cross section is equal to $<\sigma>_{Cf\text{-}252}$ = (8.952 ± 0.320) µb.

Overestimation of integral cross sections $<\sigma>_{U\text{-}235}$ in Ref. [2.22] and $<\sigma>_{Cf\text{-}252}$ in Ref. [2.23] may be explained by the fact, that data for the $^{58}Ni(n,2n)^{57}Ni$ reaction were not corrected to the recommended decay data for the ^{57}Ni from Ref. [4.62].

Evaluated excitation function for the $^{58}Ni(n,2n)^{57}Ni$ reaction was tested against the above mentioned integral experimental data [4.66] and [4.68]. Calculated averaged cross sections over ^{235}U thermal fission neutron spectrum and ^{252}Cf spontaneous fission neutron spectrum are compared with the RRDFF-2012 (ENDF/B-VII.1), JENDL/D-99 (JENDL-4.0) and experimental data in Table 4.2. Given in the last column value of C/E is the ratio of the calculated to experimental cross sections.

TABLE 4.2. CALCULATED AND MEASURED AVERAGED CROSS SECTIONS FOR THE $^{58}Ni(n,2n)^{57}Ni$ REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	Averaged cross section, μb		C/E
	Calculated	Measured	
^{235}U thermal fission neutron spectrum	3.3603 [A] 3.6332 [B] 4.2983 [C]	3.433 ± 0.230 [4.66]	0.97882 1.05832 1.25205
^{252}Cf spontaneous fission neutron spectrum	8.5428 [A] 9.2372 [B] 10.724 [C]	8.558 ± 0.310 [4.68]	0.99822 1.07936 1.25310

[A] - Present evaluation.
[B] - IRDFF-2012 (ENDF/B-VII.1).
[C] - JENDL/D-99 (JENDL-4.0).

The C/E values obtained for the ^{235}U thermal fission neutron spectrum and ^{252}Cf spontaneous fission neutron spectrum show that the $^{58}Ni(n,2n)^{57}Ni$ integral cross sections calculated from newly evaluated excitation function are agreed well with the relevant experimental data. The $<\sigma>_{U\text{-}235}$ and $<\sigma>_{Cf\text{-}252}$ cross sections calculated from the RRDFF-2012 (ENDF/B-VII.1) and JENDL/D-99 (JENDL-4.0) excitation functions agree worse with the relevant experimental values. Discrepancies between calculated and experimental values for these libraries are exceeding uncertainty in the experimental values. The $<\sigma>_{U\text{-}235}$ and $<\sigma>_{Cf\text{-}252}$ cross sections calculated from the JENDL/D-99 (JENDL-4.0) excitation function exceeds about 25% of the equivalent experimental data.

The 90%-response range for the $^{58}Ni(n,2n)^{57}Ni$ reaction in the ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are very similar in the ranges 13.10 - 18.10 MeV and 13.10 - 18.30 MeV, respectively.

The agreement between calculated and measured $<\sigma>_{U\text{-}235}$ and $<\sigma>_{Cf\text{-}252}$ cross sections confirm that the newly evaluated the $^{58}Ni(n,2n)^{57}Ni$ reaction excitation function is well determined in the energy region 13.10 - 18.30 MeV.

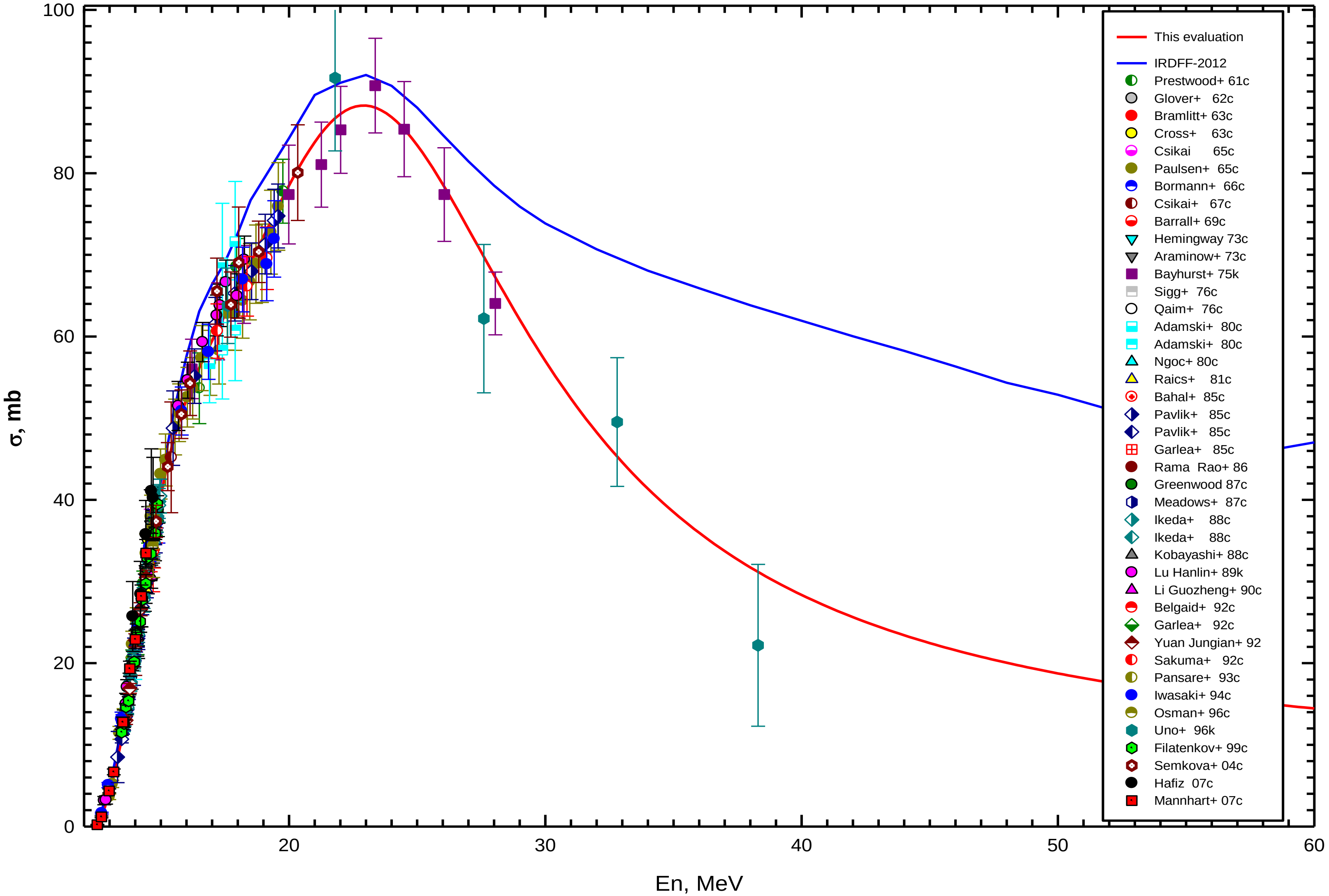

FIG. 4.1 Re-evaluated excitation function of the $^{58}Ni(n,2n)^{57}Ni$ reaction in the energy range from threshold to 60 MeV in comparison with IRDFF-2012 and corrected experimental data.

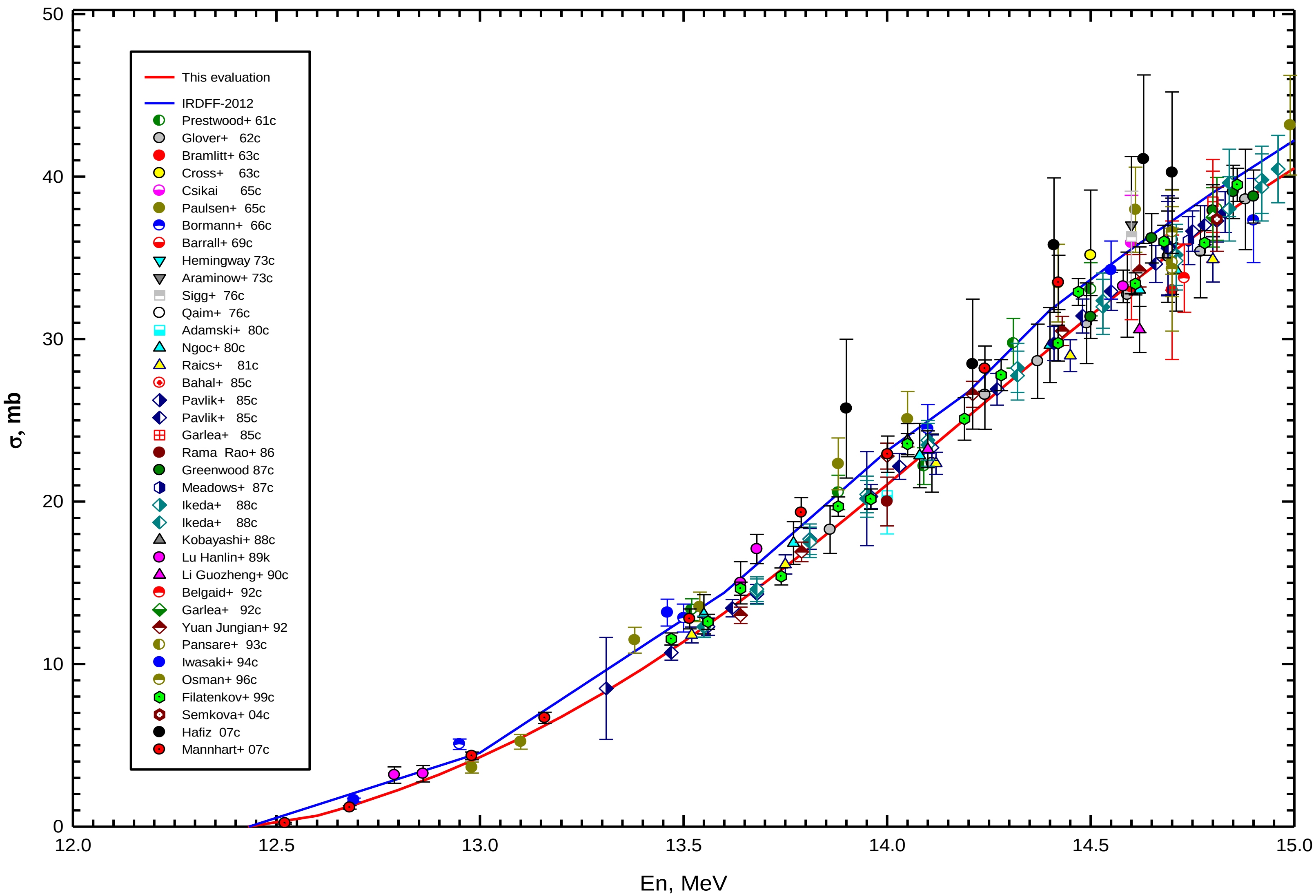

FIG. 4.2 Re-evaluated excitation function of the $^{58}Ni(n,2n)^{57}Ni$ reaction in the energy range from threshold to 15 MeV in comparison with IRDFF-2012 and corrected experimental data.

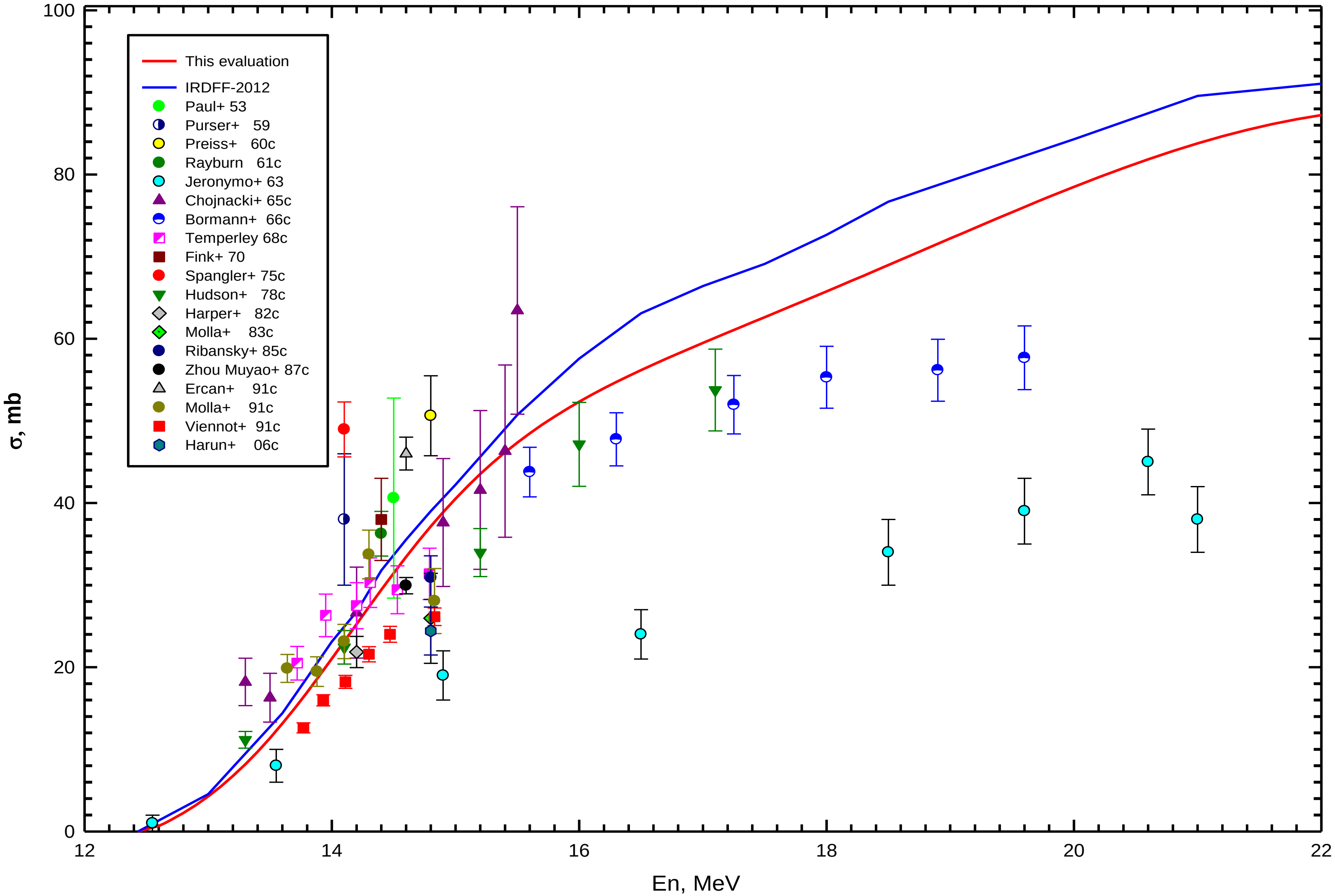

FIG. 4.3 Re-evaluated excitation function of the ^{58}Ni(n,2n)^{57}Ni reaction in the energy range from threshold to 22 MeV in comparison with IRDFF-2012 and rejected experimental data.

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5. EVALUATION OF EXCITATION FUNCTION OF THE ^{67}Zn(n,p)^{67}Cu REACTION

The isotopic abundance of ^{67}Zn in natural zinc is (4.10 ± 0.13) atom percent, and the ^{67}Cu obtained via the (n,p) reaction undergoes 100% via β^- decay with a half-life of (61.83 ± 0.12) hours. The 184.58-keV gamma radiation (I_γ = 0.487 ± 0.003) is normally used to determine the ^{67}Zn(n,p)^{67}Zn reaction rate. Recommended decay data for the half-life and gamma ray emission probabilities per decay of ^{67}Cu were taken from Ref. [2.11] of Section 2.

Experimental information about the ^{67}Zn(n,p)^{67}Cu reaction excitation function is given in the works [5.1-5.18] and covered neutron energies from 1.5 MeV to 18.2 MeV. Data of works [5.2], [5.4-5.5], [5.7-5.8], [5.10-5.18] were corrected to the new standards.

Experimental data given in the works [5.3], [5.5-5.8], [5.10], [5.11] and [5.13] were corrected also for contribution from the ^{68}Zn(n,x)^{67}Cu reaction. Data for the ^{68}Zn(n,np+pn+d)^{67}Cu excitation function were taken from a new evaluation carried out in the framework of this work [5.19]. The isotopic abundances of Zn-67 and Zn-68 in the natural Zn were taken as equal to (4.10 ± 0.13) % and (18.75 ± 0.51) %, respectively [2.13].

The database used to evaluate the ^{67}Zn(n,p)^{67}Cu reaction cross section from 1 MeV to 50 MeV was assembled from microscopic experimental data [5.1], [5.3-5.5], [5.10-5.18] and data from theoretical modeling calculations carried out by means of the modified GNASH code. Data calculated by GNASH code were renormalized to the cross section value at 16 MeV preliminarily evaluated from corrected experimental data.

Cross sections that had been determined in Refs. [5.2], [5.5-5.7] were rejected due to inconsistency with the main bulk of experimental data. For this reason were rejected also experimental data of Ghorai et al. [5.13] obtained at 14.2, 15.2, 17.2 and 18.20 MeV. Data obtained by Mirzadeh et al. for 0.0253 eV neutrons [5.9] need to be confirmed by new accurate measurements.

Evaluation of the ^{67}Zn(n,p)^{67}Cu reaction cross section and related uncertainties were performed by means of PADE-2 code. Uncertainties in the evaluated excitation function for the ^{67}Zn(n,p)^{67}Cu 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 of the relative covariance matrix in File-33 are positive:

1.80945E-06	1.83554E-06	1.87233E-06	1.91902E-06
1.99404E-06	2.06652E-06	2.16769E-06	2.29597E-06
2.40815E-06	2.60848E-06	2.81374E-06	2.81374E-06
3.41372E-06	3.77827E-06	4.17896E-06	4.79472E-06
5.32611E-06	6.00270E-06	6.97288E-06	7.63897E-06
8.97351E-06	9.96698E-06	1.14774E-05	1.30372E-05
1.47387E-05	1.68836E-05	1.90351E-05	2.15728E-05
2.47607E-05	2.71812E-05	3.22520E-05	3.41220E-05
4.03682E-05	4.46659E-05	4.90729E-05	5.81907E-05
6.16867E-05	8.56788E-05	1.23153E-04	1.83396E-04
2.86058E-04	4.76024E-04	8.73177E-04	1.88558E-03
5.53102E-03	3.09634E-02	4.30841E-02	5.67912E-02
3.59739E-01			

Evaluated group cross sections and their uncertainties for the excitation function of the ^{67}Zn(n,p)^{67}Cu reaction are listed in Table 5.1. Group boundaries are the same as in File-33.

TABLE 5.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{67}Zn(n,p)^{67}Cu$ REACTION IN THE NEUTRON ENERGY RANGE FROM 1 TO 50 MeV.

Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)	Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)
1.000 - 2.500	0.305	22.38	14.500 - 15.000	36.012	2.04
2.500 - 3.000	1.170	7.79	15.000 - 15.500	37.343	2.24
3.000 - 3.500	1.717	6.93	15.500 - 16.000	38.409	2.57
3.500 - 4.000	2.323	6.42	16.000 - 16.500	39.191	3.00
4.000 - 4.500	2.993	6.03	16.500 - 17.000	39.686	3.49
4.500 - 5.000	3.734	5.69	17.000 - 18.000	39.871	4.29
5.000 - 5.500	4.552	5.38	18.000 - 19.000	39.297	5.36
5.500 - 6.000	5.454	5.09	19.000 - 20.000	37.943	6.37
6.000 - 6.500	6.448	4.81	20.000 - 21.000	36.060	7.27
6.500 - 7.000	7.540	4.55	21.000 - 22.000	33.899	8.05
7.000 - 7.500	8.738	4.31	22.000 - 24.000	30.503	9.00
7.500 - 8.000	10.046	4.09	24.000 - 26.000	26.213	10.05
8.000 - 8.500	11.470	3.88	26.000 - 28.000	22.499	10.96
8.500 - 9.000	13.013	3.69	28.000 - 30.000	19.414	11.84
9.000 - 9.500	14.675	3.52	30.000 - 32.000	16.888	12.75
9.500 - 10.000	16.452	3.36	32.000 - 34.000	14.822	13.73
10.000 - 10.500	18.337	3.20	34.000 - 36.000	13.122	14.78
10.500 - 11.000	20.317	3.05	36.000 - 38.000	11.712	15.90
11.000 - 11.500	22.373	2.88	38.000 - 40.000	10.532	17.10
11.500 - 12.000	24.479	2.72	40.000 - 42.000	9.534	18.35
12.000 - 12.500	26.603	2.54	42.000 - 44.000	8.684	19.67
12.500 - 13.000	28.706	2.35	44.000 - 46.000	7.952	21.03
13.000 - 13.500	30.744	2.18	46.000 - 48.000	7.318	22.44
13.500 - 14.000	32.672	2.04	48.000 - 50.000	6.764	23.90
14.000 - 14.500	34.442	1.98			

Uncertainties in the evaluated $^{67}Zn(n,p)^{67}Cu$ excitation function range from 1.98% to 23.90%. The smallest uncertainties in the evaluated cross sections 1.98-2.04% are observed in the neutron energy range from 13.5 to 15.0 MeV. Significant uncertainty equal to 22.38% characterizes data in the interval 1.0 – 2.5 MeV. The highest uncertainties in cross sections exceeding 10% are in the energy range 24 - 50 MeV.

The evaluated excitation function for the $^{67}Zn(n,p)^{67}Cu$ reaction in the neutron energy range from 1 MeV to 50 MeV is shown in Fig. 5.1 in comparison with the equivalent data from TENDL-2012, EAF-2010, MENDL-2, JENDL-HE, JENDL-4.0, JEFF-3.1/A and corrected experimental data. Cross section measured by Valkonen et al. at 14.7 MeV (100.68±14.39) mb is not showed in Fig. 5.1 due to over scaling. Present evaluation in comparison with other evaluations agrees best of all with the representative experimental data. TENDL-2012 and MENDL-2 libraries in comparison with other evaluations and experimental data gives significantly overestimated cross section values in the energy ranges 8 - 15 MeV and 3 - 50 MeV, respectively.

Integral experimental data for the $^{67}Zn(n,p)^{67}Cu$ reaction are given in Refs. [5.20-5.28]. All experiments were carried out in neutron fields with a similar spectra to the ^{235}U thermal fission neutron spectrum. Experiments which have been performed in a ^{252}Cf spontaneous fission neutron spectrum are non-existant.

Measured integral cross sections for the ^{235}U thermal fission neutron spectrum extend over a wide range from (0.724 ± 0.055) mb [5.25] to (1.443 ± 0.144) mb [5.21]. Data from Ref. [5.21-5.25] and [5.27-5.28] were obtained in reactor core measurements. Two results (0.957 ± 0.107) mb [5.20] and (0.969 ± 0.063) mb [5.26] were obtained in facilities with 90%-enriched ^{235}U fission plate converter (FPC). Neutron spectra measurements show that the standard ^{235}U thermal fission neutron spectrum may be obtained at facilities with 90%-enriched ^{235}U fission plate converter with incident neutrons from a thermal column. 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. Average-weighted value obtained from FPC experimental data [5.20] and [5.26] is equal to $\langle\sigma\rangle_{U\text{-}235} = (0.966 \pm 0.054)$ mb.

Evaluated excitation function for the $^{67}Zn(n,p)^{67}Cu$ reaction was tested against the above mentioned integral experimental data. Calculated averaged cross section over ^{235}U thermal fission neutron spectrum is compared with the TENDL-2012, EAF-2010, MENDL-2, JENDL-HE, JENDL-4.0, JEFF-3.1/A and experimental data in Table 5.2. Given in the last column is the ratio of the calculated to experimental cross sections (C/E).

TABLE 5.2. CALCULATED AND MEASURED AVERAGED CROSS SECTIONS FOR THE $^{67}Zn(n,p)^{67}Cu$ REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	Averaged cross section, mb		C/E
	Calculated	Measured	
^{235}U thermal fission neutron spectrum	0.97579 [A]	0.966 ± 0.054 [*]	1.01013
	0.70247 [B]		0.72719
	1.06080 [C]		1.10567
	1.81290 [D]		1.87671
	0.83151 [E]		0.86078
	0.93673 [F]		0.96970
	0.98295 [G]		1.01755
^{252}Cf spontaneous fission neutron spectrum	1.10830 [A]		
	0.86869 [B]		
	1.23190 [C]		
	2.10780 [D]		
	0.95789 [E]		
	1.07910 [F]		
	1.10920 [G]		

[A] - Present evaluation.
[B] - TENDL-2012.
[C] - EAF-2010.
[D] - MENDL-2.
[E] - JENDL-HE.
[F] - JENDL-4.0.
[G] - JEFF-3.1/A.
[*] - Average-weighted value obtained from the FPC experimental data.

Data presented in Table 5.2 show that calculations with the evaluated $^{67}Zn(n,p)^{67}Cu$ reaction excitation function average cross section for a ^{235}U thermal fission neutron spectrum agrees within a limit of 1.0 % with experimental data. The results of calculations carried out for the JENDL-4.0 and JEFF-3.1/A libraries also agree satisfactory with experimental values. Integral cross sections calculated from MENDL-2, EAF-2010 libraries exceed experimental values by factors 1.87671 and 1.10567, respectively. Cross sections calculated from TENDL-2012, JENDL-HE excitation functions are lower than the experimental value by factors of 0.72719 and 0.86078, respectively.

The equivalent data for ^{252}Cf spontaneous fission neutron spectrum presented in Table 5.2 were determined for a relative comparison of the evaluated $^{67}Zn(n,p)^{67}Cu$ excitation functions.

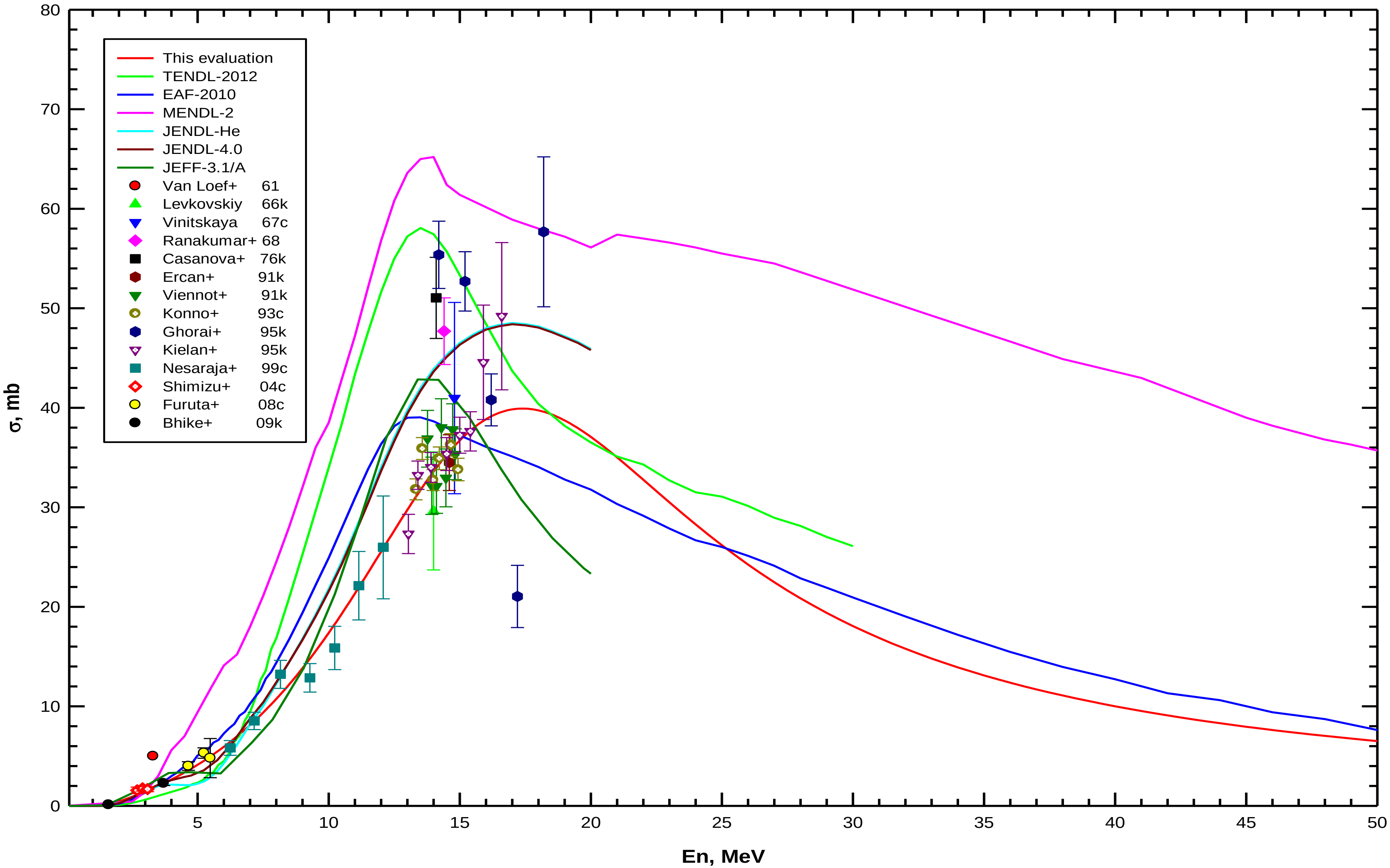

FIG. 5.1. Evaluated excitation of the ^{67}Zn(n,p)^{67}Cu reaction in the energy range from 1 MeV to 50 MeV in comparison with TENDL-2012, EAF-2010, MENDL-2, JENDL-He, JENDL-4.0, JEFF-3.1/A and experimental data.

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6. EVALUATION OF EXCITATION FUNCTION OF THE ^{92}Mo(n,p)^{92m}Nb REACTION

The isotopic abundance of ^{92}Mo in natural molybdenum is (14.84 ± 0.35) atom percent. The 135.5-keV ($J^{\pi}=2^{+}$) metastable level of ^{92}Nb exited in the (n,p) reaction undergoes 100% via ε capture and β+ decay with a half-life of (10.15 ± 0.02) days. The ε capture and β+ transition were accompanied by emission of X-ray and gamma-ray radiation. The most intense line in the gamma-ray spectrum is 934.44-keV line (I_{γ} = 0.9907 ± 0.0004). Recommended decay data for the half-life and gamma ray emission probability per decay of ^{92m}Nb were taken from Ref. [2.11] of Section 2.

Experimental information about the ^{92}Mo(n,p)^{92m}Nb reaction excitation function is given in the works [6.1-6.35] and covered neutron energies range from 1.6 MeV to 20.5 MeV. Experimental data of works [6.1-6.3], [6.5-6.7], [6.9-6.18] and [6.20-6.35] in the process of analysis were corrected on the basis of the newly recommended cross section data for the relevant monitor reactions and the recommended decay data (see Table 2.1).

The database used to evaluate the ^{92}Mo(n,p)^{92m}Nb reaction cross section from 1 MeV to 40 MeV was assembled from microscopic experimental data [6.1], [6.3], [6.6-6.17], [6.20], [6.22-6.29], [6.31], [6.32], [6.34-6.35] and data from theoretical modeling calculation carried out by means of modified GNASH code. Experimental data of Reimer et al. obtained in the energy range 16.2 - 20.5 MeV were taken from the last publication [6.32]. Data calculated by GNASH code were renormalized to the cross section value determined in measurements of Zhao Wenrong et al. at 19.09 MeV [6.25].

Cross sections that had been measured in works [6.2], [6.4-6.5], [6.19], [6.21], [6.33] were rejected due to inconsistency with the main bulk of experimental data. Experimental data of Liskien et al. [6.19] agree well at 12.59 MeV (109.9 ± 5.6 mb) with data by Doczi et al [6.26] at 12.53 MeV (109.52 ± 6.47 mb), but with increase of the incident neutron energy discrepancy between the representative experimental data and data obtained by Liskien et al. is significantly growing.

Evaluation of the excitation function of the ^{92}Mo(n,p)^{92m}Nb reaction from 1 MeV 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 ^{92}Mo(n,p)^{92m}Nb reaction are given in the form of a relative covariance matrix for 49-neutron energy groups (LB = 5). Covariance matrix of uncertainties was calculated simultaneously with the recommended cross section data by means of the PADE-2 code and tested in additional by COVEIG code [3.73]. Six-digit eigenvalues of the relative covariance matrix in File-33 are as follows:

```
6.69193E-06    7.28092E-06    8.09184E-06    9.04294E-06
1.01202E-05    1.13789E-05    1.28633E-05    1.49382E-05
1.74953E-05    2.10032E-05    2.47084E-05    3.02532E-05
3.60682E-05    4.17556E-05    4.66232E-05    5.21772E-05
6.03931E-05    6.44122E-05    6.92972E-05    7.73120E-05
8.42908E-05    9.25466E-05    1.06442E-04    1.24425E-04
1.27774E-04    1.44328E-04    1.61751E-04    1.78296E-04
1.94021E-04    2.09027E-04    2.23394E-04    2.37120E-04
2.49813E-04    2.59087E-04    2.72743E-04    2.97448E-04
3.20078E-04    3.34536E-04    3.55338E-04    7.99532E-04
1.12947E-03    3.12196E-03    4.71536E-03    7.91728E-03
9.27525E-03    1.51048E-02    3.44107E-02    6.36581E-02
2.55476E-01
```

All of these eigenvalues are positive.

Evaluated group cross sections and their uncertainties for the excitation function of the ^{92}Mo(n,p)^{92m}Nb reaction are listed in Table 6.1. Group boundaries are the same as in File-33.

TABLE 6.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE ^{92}Mo(n,p)^{92m}Nb REACTION IN THE NEUTRON ENERGY RANGE FROM 1 TO 50 MeV.

Neutron energy (MeV) from to	**Cross section (mb)**	**Uncer-tainty (%)**	**Neutron energy (MeV) from to**	**Cross section (mb)**	**Uncer-tainty (%)**
1.000 - 3.000	1.202	25.12	15.000 - 15.500	53.661	1.82
3.000 - 3.500	7.539	7.71	15.500 - 16.000	48.284	2.31
3.500 - 4.000	14.079	6.71	16.000 - 16.500	43.893	2.82
4.000 - 4.500	23.314	6.07	16.500 - 17.000	40.303	3.31
4.500 - 5.000	34.853	5.75	17.000 - 17.500	37.353	3.77
5.000 - 5.500	47.504	5.45	17.500 - 18.000	34.911	4.20
5.500 - 6.000	59.678	5.00	18.000 - 18.500	32.871	4.59
6.000 - 6.500	70.133	4.55	18.500 - 19.000	31.152	4.95
6.500 - 7.000	78.436	4.27	19.000 - 20.000	29.060	5.45
7.000 - 7.500	84.869	4.15	20.000 - 21.000	26.874	6.07
7.500 - 8.000	90.052	4.04	21.000 - 22.000	25.179	6.68
8.000 - 8.500	94.615	3.85	22.000 - 23.000	23.833	7.31
8.500 - 9.000	99.030	3.59	23.000 - 24.000	22.733	8.00
9.000 - 9.500	103.537	3.30	24.000 - 25.000	21.813	8.75
9.500 - 10.000	108.092	3.00	25.000 - 26.000	21.028	9.57
10.000 - 10.500	112.337	2.69	26.000 - 27.000	20.347	10.45
10.500 - 11.000	115.647	2.44	27.000 - 28.000	19.746	11.39
11.000 - 11.500	117.466	2.34	28.000 - 29.000	19.210	12.37
11.500 - 12.000	118.082	2.28	29.000 - 30.000	18.727	13.38
12.000 - 12.500	115.619	2.32	30.000 - 32.000	18.086	14.95
12.500 - 13.000	102.196	2.40	32.000 - 34.000	17.338	17.10
13.000 - 13.500	88.386	1.84	34.000 - 36.000	16.690	19.28
13.500 - 14.000	77.429	1.44	36.000 - 38.000	16.117	21.45
14.000 - 14.500	68.106	1.30	38.000 - 40.000	15.602	23.60
14.500 - 15.000	60.216	1.44			

Uncertainties in the evaluated ^{92}Mo(n,p)^{92m}Nb excitation function range from 1.30% to 25.12%. The smallest uncertainties in the evaluated cross sections 1.30-1.84% are observed in the neutron energy range from 13.0 to 15.5 MeV. The highest uncertainty equal to 25.12% characterize data in the interval 1.0 – 3.0 MeV. Significant uncertainties in cross sections exceeding 10% are in the energy range 26 - 40 MeV.

Fig. 6.1 shows the evaluated excitation function for the ^{92}Mo(n,p)^{92m}Nb reaction over the neutron energy range from 1 – 40.0 MeV in comparison with EAF-2010, JEFF-3.1/A and corrected experimental data. The evaluated excitation function for the ^{92}Mo(n,p)^{92m}Nb reaction in the neutron energy range 11 – 16 MeV in comparison with EAF-2010, JEFF-3.1/A and corrected experimental data are presented in Fig. 6.2. The same information in comparison with rejected experimental data are presented in Fig. 6.3.

Comparison of excitation functions show that EAF-2010, JEFF-3.1/A and the new evaluation agree satisfactorily only in the energy interval 12.5 – 14.5 MeV. Below this energy range the EAF-2010 and JEFF-3.1/A evaluations gives systematically higher cross sections, while above 14.5 MeV cross sections are systematically lower than the present evaluation.

Integral experiments for the $^{92}Mo(n,p)^{92m}Nb$ reaction are described in the works [6.36-6.47]. Eleven experiments were carried out in the neutron fields with spectra which are similar to ^{235}U thermal fission neutron spectrum [6.36-6.46] and only one experiment [6.47] was performed in the field of ^{252}Cf spontaneous fission neutron spectrum. Experimental data obtained for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were corrected to the new recommended cross sections for monitor reactions and decay data.

Measured integral cross sections for ^{235}U thermal fission neutron spectrum have a range of obtained results from (3.751 ± 0. 375) mb [6.36] to (7.061 ± 0.455) mb [6.37]. Data from Ref. [6.36], [6.40-6.44] and [6.46] were obtained in reactor core measurements. Five results (7.061 ± 0.455) mb [6.37], (6.759 ± 0.341) mb [6.38], (6.483 ± 0.294) mb [6.39], (6.505 ± 0.474) mb [6.42], (6.747 ± 0.427) mb [6.45] were determined in facilities with 90%-enriched ^{235}U fission plate converter (FPC).

The results of reactor core measurements were not taken into account in calculation of average-weighted cross section value for the $^{92}Mo(n,p)^{92m}Nb$ reaction. Average-weighted value obtained from FPC experimental data [6.37-6.39], [6.42] and [6.45] is equal to $\langle\sigma\rangle_{U\text{-}235}$ = (6.687 ± 0.169) mb.

Evaluated excitation function for the $^{92}Mo(n,p)^{92m}Nb$ reaction was tested against the above mentioned integral experimental data. Calculated averaged cross sections over ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are compared with the equvalent EAF-2010, JEFF-3.1/A and experimental data in Table 6.2. Given in the last column is the ratio of the calculated to experimental cross sections (C/E).

TABLE 6.2. CALCULATED AND MEASURED AVERAGED CROSS SECTIONS FOR THE $^{92}Mo(n,p)^{92m}Nb$ REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	**Averaged cross section, mb**		**C/E**
	Calculated	**Measured**	
^{235}U thermal fission neutron spectrum	6.7337 [A] 9.1825 [B] 8.9462 [C]	6.687 ± 0.169 [*]	1.00698 1.37319 1.33785
^{252}Cf spontaneous fission neutron spectrum	7.8349 [A] 10.5200 [B] 10.2610 [C]	15.170 ± 0.667 [6.47]	0.51242 0.69347 0.67640

[A] - Present evaluation.
[B] - EAF-2010.
[C] - JEFF-3.1/A.
[*] - Average-weighted value obtained from the FPC experimental data.

Data presented in Table 6.2 show that calculated ^{235}U thermal fission neutron spectrum average cross section from the evaluated $^{92}Mo(n,p)^{92m}Nb$ reaction excitation function agrees best of all with experimental data. The results of calculations carried out for the EAF-2010 and JEFF-3.1/A libraries significantly exceed the evaluated experimental value. Integral cross section values calculated from all evaluations for ^{252}Cf spontaneous fission neutron spectrum are significantly lower in comparison with experimental data of Z. Dezso and J. Csikai [6.47]. This fact permits us to conclude that new precise measurements in the ^{252}Cf spontaneous fission neutron spectrum are desirable for the $^{92}Mo(n,p)^{92m}Nb$ reaction.

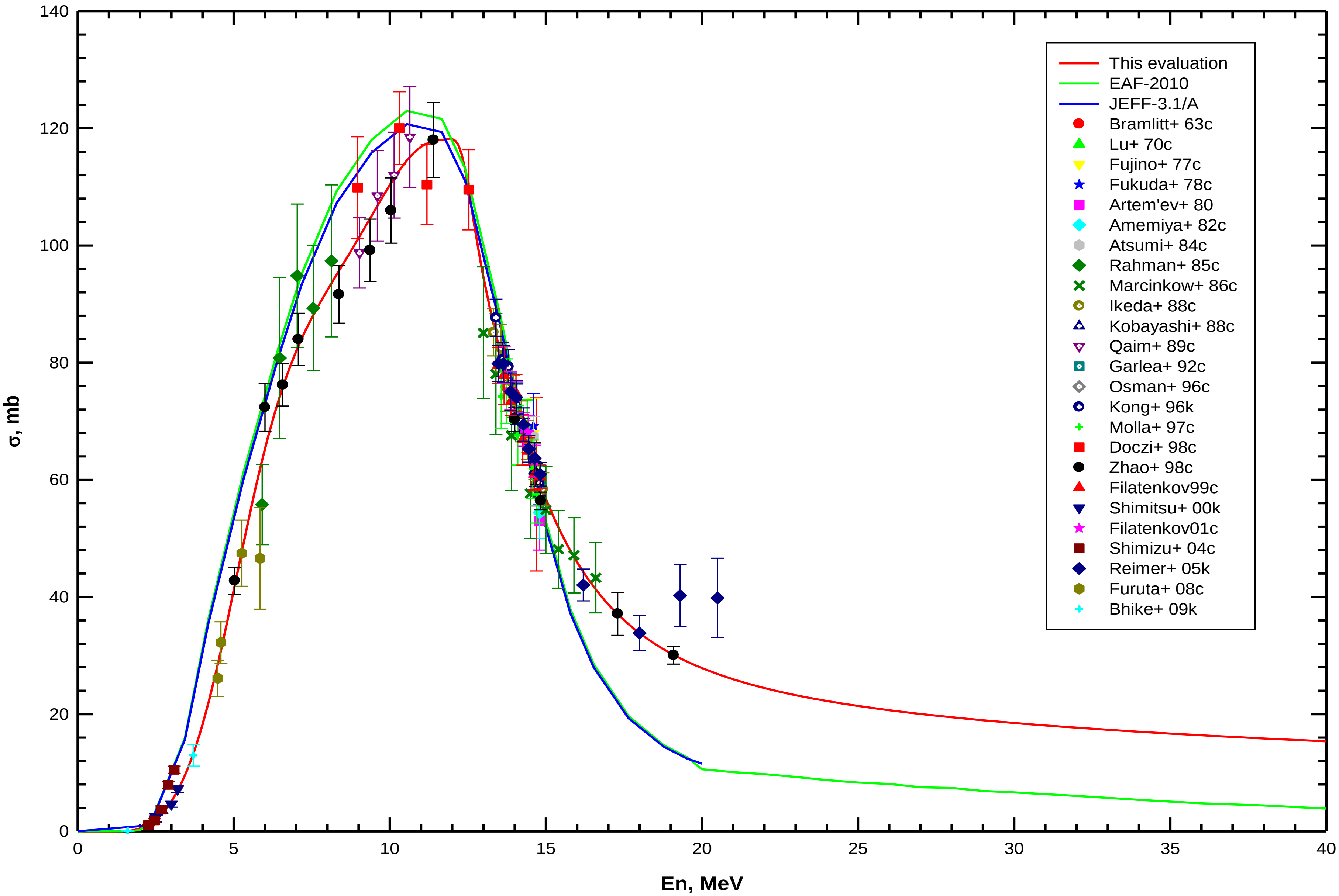

FIG. 6.1. Evaluated excitation function of the ^{92}Mo(n,p)^{92m}Nb reaction in the energy range (1 – 40) MeV in comparison with TENDL-2010, JEFF-3.1/A and corrected experimental data.

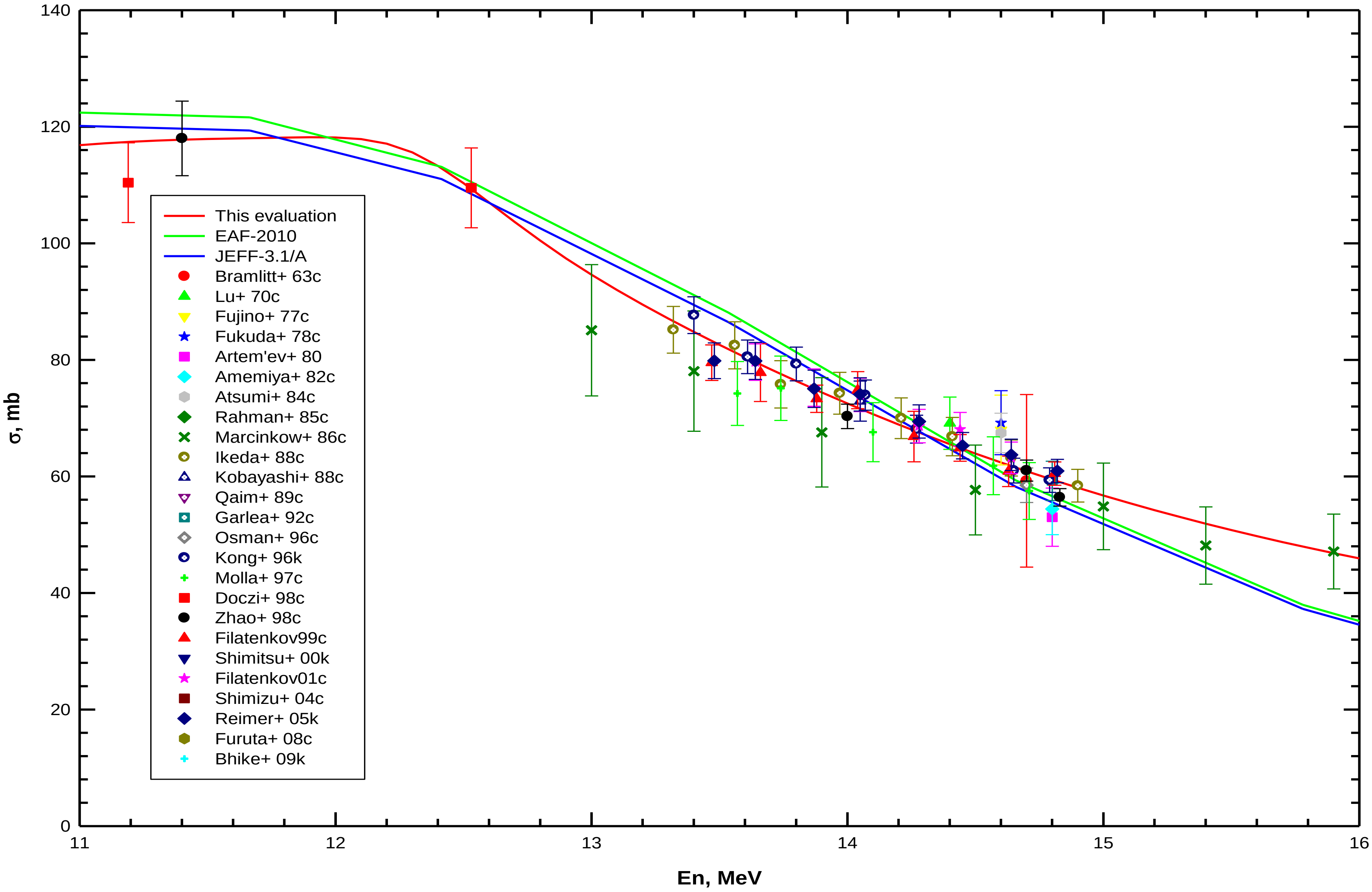

FIG. 6.2. Evaluated excitation function of the ^{92}Mo(n,p)^{92m}Nb reaction in the energy range (11 – 16) MeV in comparison with TENDL-2010, JEFF-3.1/A and corrected experimental data.

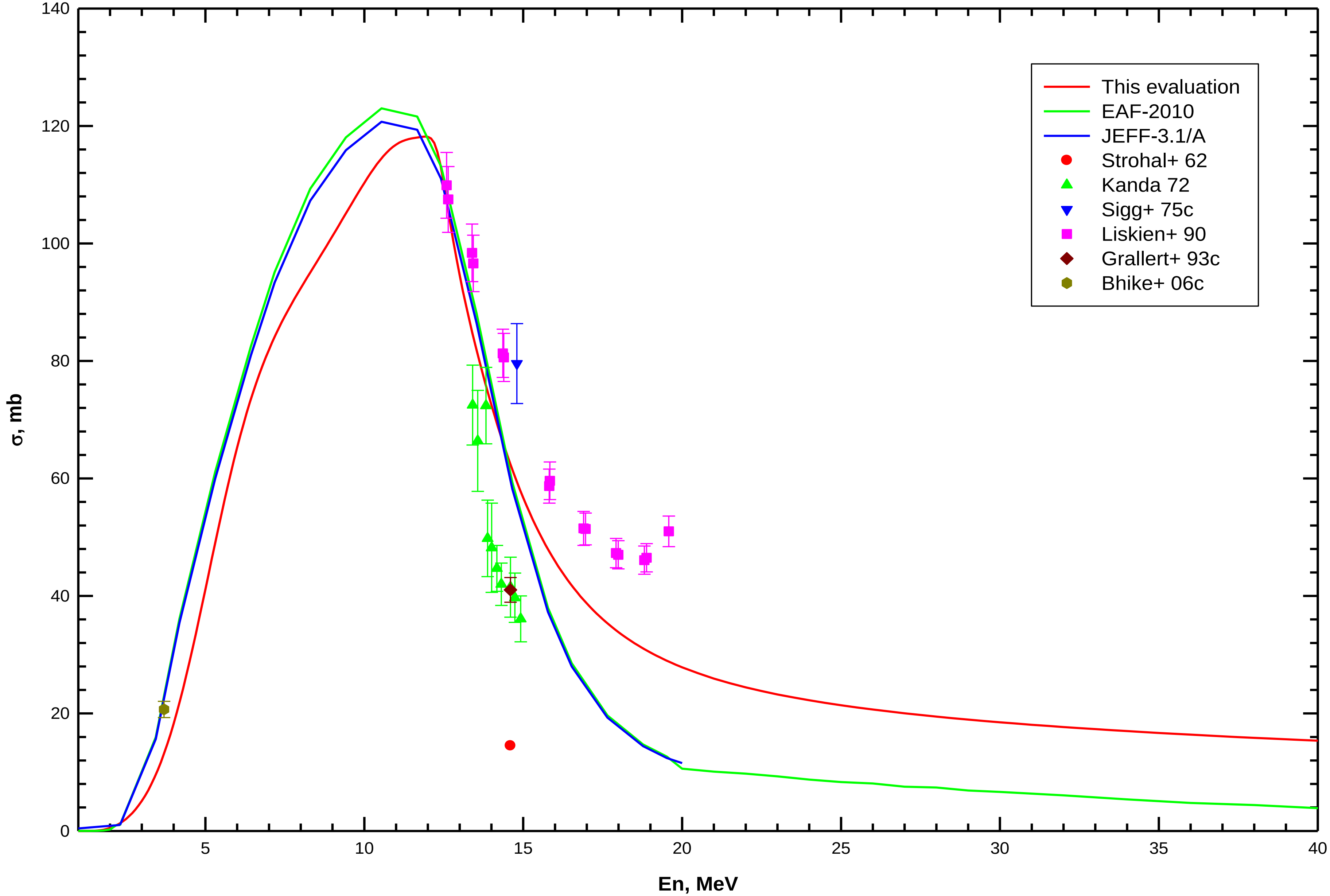

FIG. 6.3. Evaluated excitation function of the ^{92}*Mo(n,p)*^{92m}*Nb reaction in the energy range (1 – 40) MeV in comparison with EAF-2010, JEFF-3.1/A and rejected experimental data.*

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7. EVALUATION OF EXCITATION FUNCTION OF THE ^{93}Nb(n,γ)^{94}Nb REACTION

The abundance of ^{93}Nb isotope in natural indium is 100 atom percent. The ($J^{\pi}=6^{+}$) ground level of ^{94}Nb populated by the (n,γ) reaction undergoes 100% via β^{-} decay with a half-life of (20300 ± 16) years. The β^{-} transition is accompanying with emission of X-ray and gamma-ray radiation. The most intense lines in gamma-ray spectrum are 702.65 keV line with intensity I_{γ} = 0.99814 and 871.091 keV line I_{γ} = 0.99892. Recommended decay data for the half-life and gamma ray emission probability per decay of ^{94}Nb were taken from Ref. [7.32].

Microscopic experimental data [7.1-7.31] were analysed in order to evaluate the ^{93}Nb(n,γ)^{94}Nb reaction excitation function. During this procedure experimental data [7.5-7.7], [7.9-7.10], [7.13], [7.15], [7.18], [7.22-7.25], [7.27-7.30] were corrected using new standards for the relevant monitor reactions and the recommended decay data (see Table 2.1). Gibbons et al. used as the reference cross section data for the In-nat(n,γ) reaction. New recommended cross sections for this reaction were calculated from the ENDF/B-VII.1 data for ^{113}In(n,γ) and ^{115}In(n,γ) reactions. Original relative experimental data of Stavisskij and Shapar' [7.10] were normalized by authors to the cross section value of 65 mb at 0.4 MeV point. These experimental data were re-normalized to the preliminary evaluated cross section value of (55.9 ± 2.5) mb for neutron energy 0.4 MeV.

The results of measurements of the ^{93}Nb(n,γ)^{94}Nb reaction cross section for thermal neutrons are presented in works [7.1-7.3], [7.8], [7.14] and [7.16]. The more representative experimental data are given in publications [7.8] and [7.14]. These data were used apparently by S.F. Mughabghab to obtain the recommended cross section of (1.15 ± 0.05) b at 0.0253 eV [7.34]. This cross section value at 0.0253 eV was adopted in the evaluation of the ^{93}Nb(n,γ)^{94}Nb reaction excitation function.

Microscopic experimental data for the ^{93}Nb radiative capture cross section above 3 MeV are very poor. We have today only one experiment by F. Rigaud et al. [7.17]. In this experiment the authors had measured the ^{93}Nb(n,γ)^{94}Nb reaction cross section at the neutron energy 14.06 MeV. Therefore in the evaluation of the ^{93}Nb(n,γ)^{94}Nb reaction excitation function above 3 MeV the dominant data were obtained from theoretical model calculation performed by means of the modified GNASH code.

In describing the new evaluation the ^{93}Nb(n,γ)^{94}Nb reaction cross section in the energy range 1.0E-05 eV – 7.5 keV is given via the Reich-Moore resonance parameters. Resonance parameters evaluated by S.F. Mughabghab [2.21] were used as preliminary source of information. The work [2.21] presents information about 203 resonances of ^{93}Nb. Parameters of 40 resonances are completely identified. Values of L or J^{π} parameters for 35 resonances are questionable. All the remaining resonances are described in work [2.21] with an incomplete set of parameters.

For determination of the ^{93}Nb resonance parameters all available experimental data carried out for measurement of parameters and cross sections for the ^{93}Nb(n,γ), ^{93}Nb(n,tot) and ^{93}Nb(n,el) reactions in resolved resonance region were analysed. Experimental data for total neutron cross section on ^{93}Nb in resolved resonance region were taken from the works [7.34-7.42]. The results of capture cross section measurements are published in the works [7.11], [7.13], [7.28-7.30]. Information presented in the work [7.29] is repeated in publication [7.30]. Unfortunately, experimental cross section data for the ^{93}Nb(n,el) reaction in the resonance energy range are given only in one work [7.33]. Neutron elastic cross sections for ^{93}Nb were measured by authors of this work in the energy range 1.80 – 442 keV.

In the process of evaluation the resonance parameter re-constructed cross sections of the ^{93}Nb(n,tot), ^{93}Nb(n,γ)^{94}Nb and ^{93}Nb(n,el) reactions were compared with corresponding experimental data. The tested values of the resonance parametrs were: E_r – energy of resonance, L – neutron orbital moment, J^π – spin and parity of resonance, Γ_n – neutron width, Γ_γ – radiative width.

The more representative for testing resonance parameters were results obtained in high resolution measurements [7.35], [7.37] and [7.42]. Neutron radiative capture cross section in the resonance region was tested by comparison with averaged over intervals experimental data [7.13], [7.28] and [7.30].

As a result of the analysis, parameters for 70 s-resonances and 180 p-resonances were determined. In total parameters for 250 resonances were evaluated. The obtained resonance parameters permit to calculate the ^{93}Nb(n,tot), ^{93}Nb(n,el) and ^{93}Nb(n,γ)^{94}Nb excitation functions without any additional data (background) in the neutron energy range 1.0E-05 eV – 7.5 keV.

Some results of testing the evaluated Reich-Moore resonance parameters are shown in Figs. 7.1-7.4 in comparison with equivalent data from the ENDF/B-VII.1 library and experimental data. Parameters of the lowest resonance excited in the compound system n + ^{93}Nb were adopted in the new evaluation as follows: E_r = 35.89 eV, L= 1, J^π = 5, Γ_n = 1.430E-04 eV, Γ_γ = 2.280E-01 eV. Values of Γ_n and Γ_γ were taken from measurements by Grigoriev et al. [7.28]. The ^{93}Nb(n,tot) cross section calculated with these parameters agrees well with microscopic cross sections measured by Saplakoglu et al. in the region of this resonance (See Fig. 7.1). Total ^{93}Nb cross section re-constructed from the evaluated Reich-Moore resonance parameters in the energy interval 4 – 7.5 keV is shown on Figs. 7.2, 7.3 and 7.4. As is seen in all Figs. the ^{93}Nb(n,tot) excitation function obtained from new resonance parameters agree better with microscopic cross sections measured in the energy range 4 - 7.5 keV when compared with equivalent values from ENDF/B-VII.1 library.

Calculated from the evaluated Reich-Moore resonance parameters the ^{93}Nb(n,el) and ^{93}Nb(n,γ)^{94}Nb reaction cross sections at E_n = 0.0253 eV and T = 293K are equal to 6.37543 barns and 1.15026 barns, respectively. Both values agree well with equivalent cross sections (6.37 ± 0.07) b and (1.15 ± 0.05) b recommended by Mughabghab.

Evaluation of the excitation function of the ^{93}Nb(n,γ)^{94}Nb reaction from 7.5 keV to 20 MeV was carried out by means of the generalized least-squares method within the PADE-2 code. The database used to evaluate the ^{93}Nb(n,γ)^{94}Nb reaction cross section in this energy range was assembled from microscopic experimental data described above and data from theoretical modeling calculation carried out by means of the modified GNASH code.

Uncertainties in the evaluated ^{93}Nb(n,γ)^{94}Nb reaction excitation function are given via two independent matrices.

In the resolved resonance region 1.0E-5 - 7.5 keV uncertainties are presented in the form of a relative covariance matrix for the 37-neutron energy groups (LB = 5). In the energy range 7.5 keV - 20 MeV uncertainties are given also in the form of relative covariance matrix for the 47-neutron energy groups (LB = 5). Eigenvalues of the both covariance matrix given in File-33 were calculated by means of PADE-2 and tested additionally by the COVEIG code [3.73]. It was found that all eigenvalues are positive.

Evaluated group cross sections and related uncertainties for the ^{93}Nb(n,γ)^{94}Nb reaction excitation function are listed in Table 7.1. Group boundaries are the same as in File-33.

TABLE 7.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{93}Nb(n,\gamma)^{94}Nb$ REACTION IN THE NEUTRON ENERGY RANGE FROM 1.0E-5 eV TO 20 MeV.

Neutron energy (eV) from - to	Cross section (b)	Uncer-tainty (%)	Neutron energy (eV) from - to	Cross section (b)	Uncer-tainty (%)
1.00E-5 - 1.00E-4	2.78015E+1	6.80	2.00E+4 - 2.40E+4	3.48567E-1	0.96
1.00E-4 - 1.00E-3	8.79146E+0	6.23	2.40E+4 - 2.90E+4	3.03769E-1	0.87
1.00E-3 - 1.00E-2	2.78007E+0	5.84	2.90E+4 - 3.50E+4	2.62877E-1	0.82
1.00E-2 - 2.00E-2	1.51562E+0	4.34	3.50E+4 - 4.10E+4	2.29659E-1	0.81
2.00E-2 - 3.00E-2	1.16281E+0	3.81	4.10E+4 - 4.90E+4	2.00855E-1	0.81
3.00E-2 - 5.00E-2	9.21895E-1	4.00	4.90E+4 - 5.90E+4	1.73577E-1	0.81
5.00E-2 - 7.00E-2	7.49122E-1	4.71	5.90E+4 - 7.00E+4	1.50548E-1	0.81
7.00E-2 - 1.00E-1	6.29584E-1	5.28	7.00E+4 - 8.40E+4	1.30960E-1	0.82
1.00E-1 - 1.60E-1	5.10210E-1	6.14	8.40E+4 - 1.00E+5	1.14178E-1	0.88
1.60E-1 - 2.50E-1	4.05719E-1	7.44	1.00E+5 - 1.20E+5	1.00092E-1	0.99
2.50E-1 - 4.00E-1	3.22007E-1	7.80	1.20E+5 - 1.40E+5	8.89055E-2	1.13
4.00E-1 - 6.30E-1	2.55142E-1	7.65	1.40E+5 - 1.70E+5	7.91897E-2	1.30
6.30E-1 - 1.00E+0	2.02198E-1	7.05	1.70E+5 - 2.00E+5	7.11532E-2	1.50
1.00E+0 - 1.60E+0	1.59305E-1	6.60	2.00E+5 - 2.50E+5	6.42519E-2	1.79
1.60E+0 - 2.50E+0	1.25721E-1	6.42	2.50E+5 - 3.00E+5	5.88462E-2	2.20
2.50E+0 - 4.00E+0	9.86517E-2	6.65	3.00E+5 - 4.00E+5	5.46939E-2	2.73
4.00E+0 - 6.30E+0	7.67954E-2	7.39	4.00E+5 - 5.00E+5	5.23733E-2	3.15
6.30E+0 - 1.00E+1	5.92833E-2	9.14	5.00E+5 - 6.00E+5	5.16098E-2	3.27
1.00E+1 - 1.60E+1	4.49780E-2	13.16	6.00E+5 - 8.00E+5	4.95966E-2	3.38
1.60E+1 - 2.00E+1	3.66908E-2	15.70	8.00E+5 - 1.00E+6	4.12047E-2	3.42
2.00E+1 - 3.00E+1	3.21319E-2	12.54	1.00E+6 - 1.20E+6	2.91820E-2	3.95
3.00E+1 - 4.00E+1	9.28967E-1	13.05	1.20E+6 - 1.50E+6	1.91466E-2	5.11
4.00E+1 - 6.00E+1	2.38917E-1	12.18	1.50E+6 - 1.80E+6	1.40044E-2	5.68
6.00E+1 - 8.00E+1	1.42725E-2	11.43	1.80E+6 - 2.00E+6	1.16797E-2	5.51
8.00E+1 - 1.00E+2	3.95698E-1	8.81	2.00E+6 - 2.50E+6	8.52233E-3	5.71
1.00E+2 - 2.00E+2	3.60146E+0	5.66	2.50E+6 - 3.00E+6	4.69401E-3	6.75
2.00E+2 - 3.00E+2	1.91620E-1	4.94	3.00E+6 - 3.50E+6	2.45590E-3	6.99
3.00E+2 - 4.00E+2	3.59614E+0	4.76	3.50E+6 - 4.00E+6	1.31721E-3	7.26
4.00E+2 - 6.00E+2	2.79264E-1	4.70	4.00E+6 - 5.00E+6	5.92003E-4	8.34
6.00E+2 - 8.00E+2	1.51518E+0	4.65	5.00E+6 - 6.00E+6	2.62705E-4	11.99
8.00E+2 - 1.00E+3	1.47257E+0	4.59	6.00E+6 - 7.00E+6	2.20582E-4	13.49
1.00E+3 - 2.00E+3	1.94558E+0	4.38	7.00E+6 - 8.00E+6	2.88285E-4	11.80
2.00E+3 - 3.00E+3	1.52796E+0	4.02	8.00E+6 - 9.00E+6	4.07581E-4	10.53
3.00E+3 - 4.00E+3	9.06170E-1	3.70	9.00E+6 - 1.00E+7	5.51057E-4	10.10
4.00E+3 - 5.00E+3	7.53904E-1	3.42	1.00E+7 - 1.10E+7	6.97290E-4	9.99
5.00E+3 - 6.00E+3	8.92467E-1	3.18	1.10E+7 - 1.20E+7	8.21449E-4	9.90
6.00E+3 - 7.50E+3	7.12043E-1	2.93	1.20E+7 - 1.30E+7	8.97414E-4	9.82
7.50E+3 - 1.00E+4	6.23839E-1	1.66	1.30E+7 - 1.40E+7	9.08593E-4	9.87
1.00E+4 - 1.20E+4	5.49618E-1	1.40	1.40E+7 - 1.50E+7	8.56243E-4	10.08
1.20E+4 - 1.40E+4	4.96926E-1	1.27	1.50E+7 - 1.60E+7	7.58636E-4	10.39
1.40E+4 - 1.70E+4	4.44266E-1	1.16	1.60E+7 - 1.80E+7	5.79910E-4	11.17
1.70E+4 - 2.00E+4	3.94060E-1	1.06	1.80E+7 - 2.00E+7	3.61523E-4	13.00

Uncertainties in the evaluated $^{93}Nb(n,\gamma)^{94}Nb$ excitation function range from 0.81 to 15.70%. The smallest uncertainties lower than 2% are observed in the neutron energy range from 7.5 keV to

0.25 MeV. Uncertainties in cross sections exceed 10% in the energy ranges 10 - 80 eV, 5 - 10 MeV and 14 - 20 MeV.

The evaluated excitation function for the $^{93}Nb(n,\gamma)^{94}Nb$ reaction is displayed in Figs. 7.5, 7.6 and 7.7 for three energy regions: 1.0E-10 – 20 MeV, 0.00001 – 0.01 MeV and 0.01 – 20 MeV, respectively, in comparison with IRDFF-2012 (ENDF/B-VII.1) and experimental data. All evaluated data are plotted in Figs. in the SAND-II 640-energy groups.

Integral experiments for the $^{93}Nb(n,\gamma)^{94}Nb$ reaction which may be used in benchmark calculations for testing evaluated microscopic cross section are non existent.

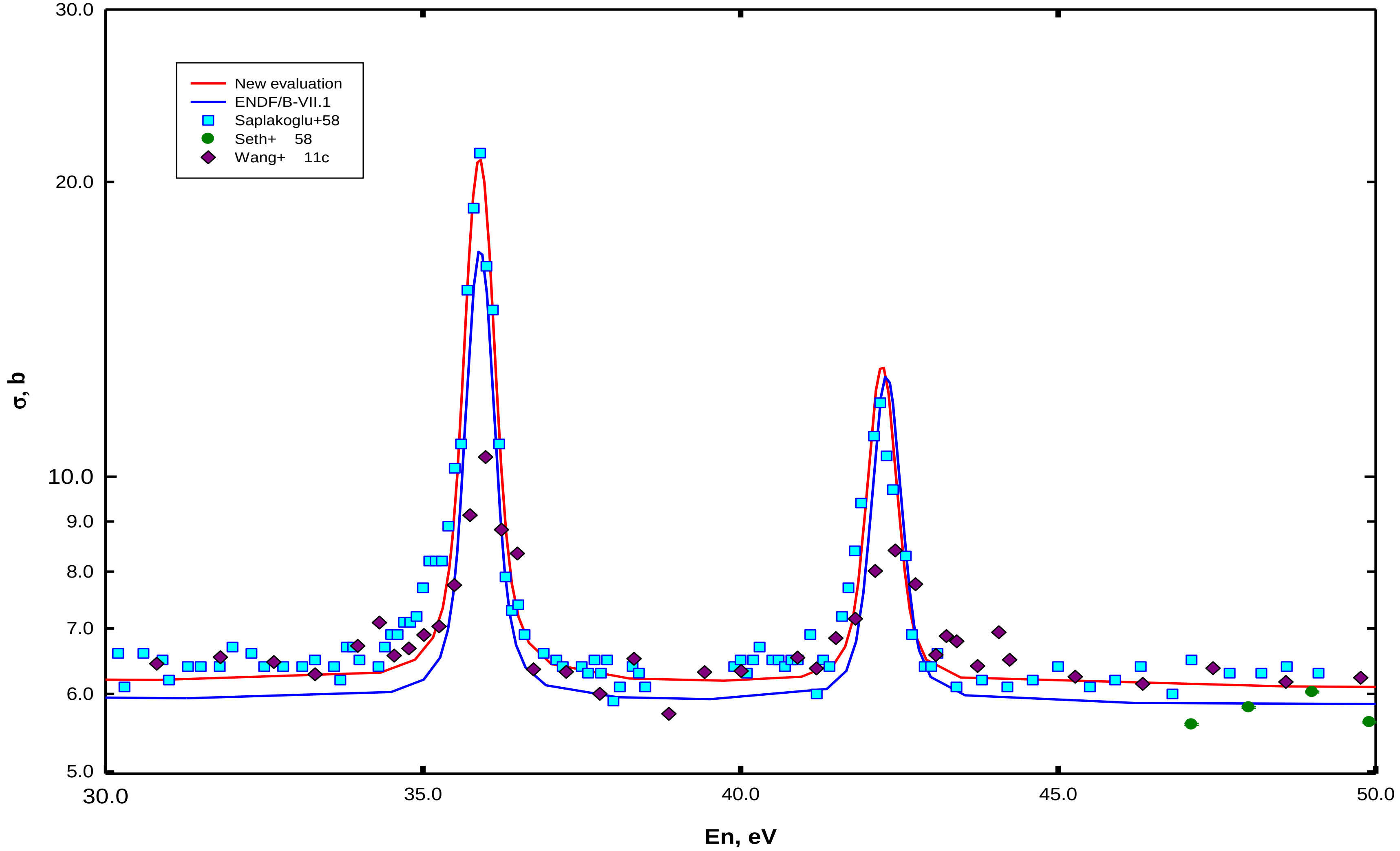

FIG. 7.1. Reconstructed from the evaluated resonance parameters ^{93}Nb(n,tot) cross sections in the energy range 30-50 eV in comparison with equivalent ENDB/B-VII.1 and experimental data.

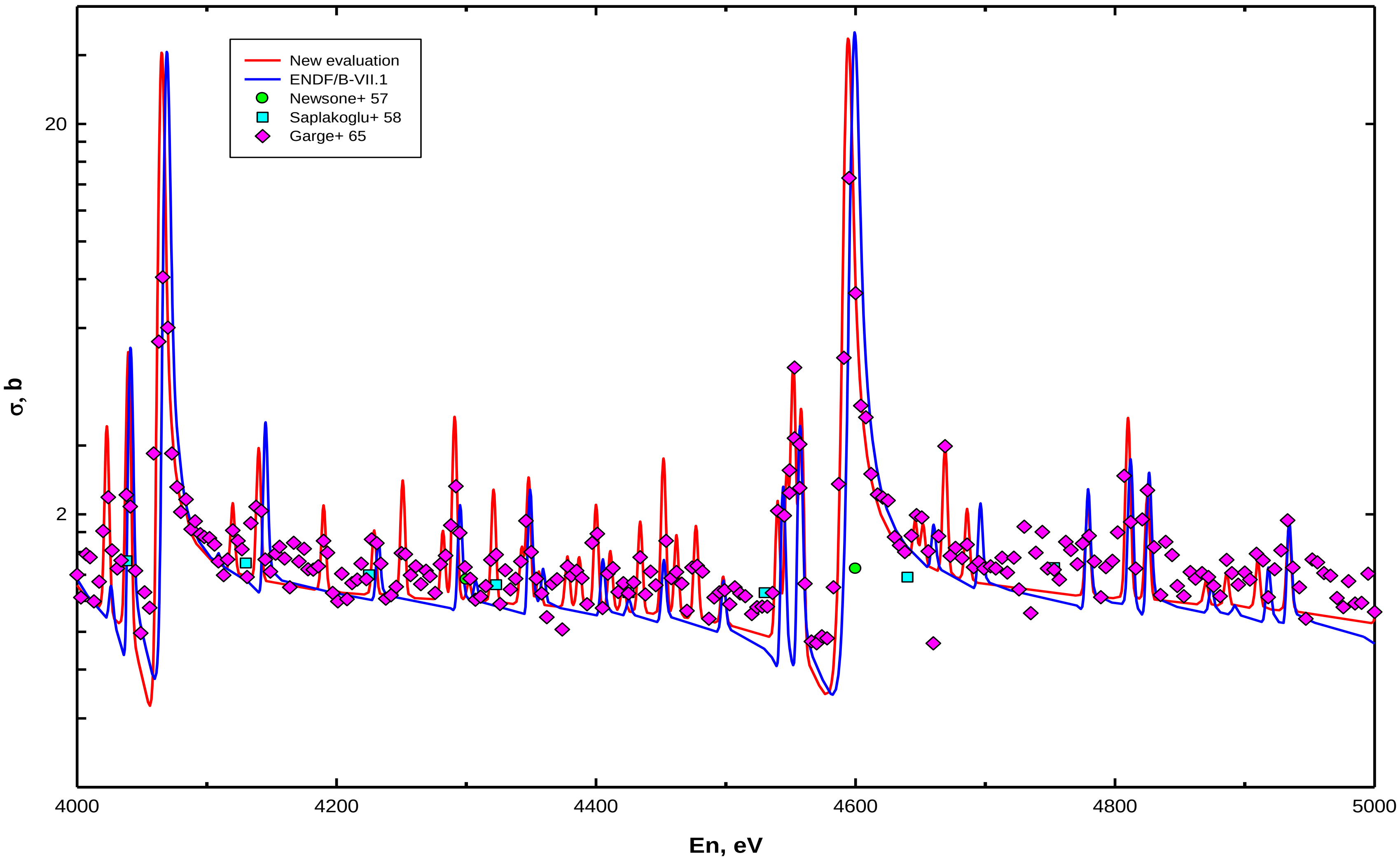

FIG. 7.2. Reconstructed from the evaluated resonance parameters ^{93}Nb(n,tot) cross sections in the energy range 4000-5000 eV in comparison with equivalent ENDB/B-VII.1 and experimental data.

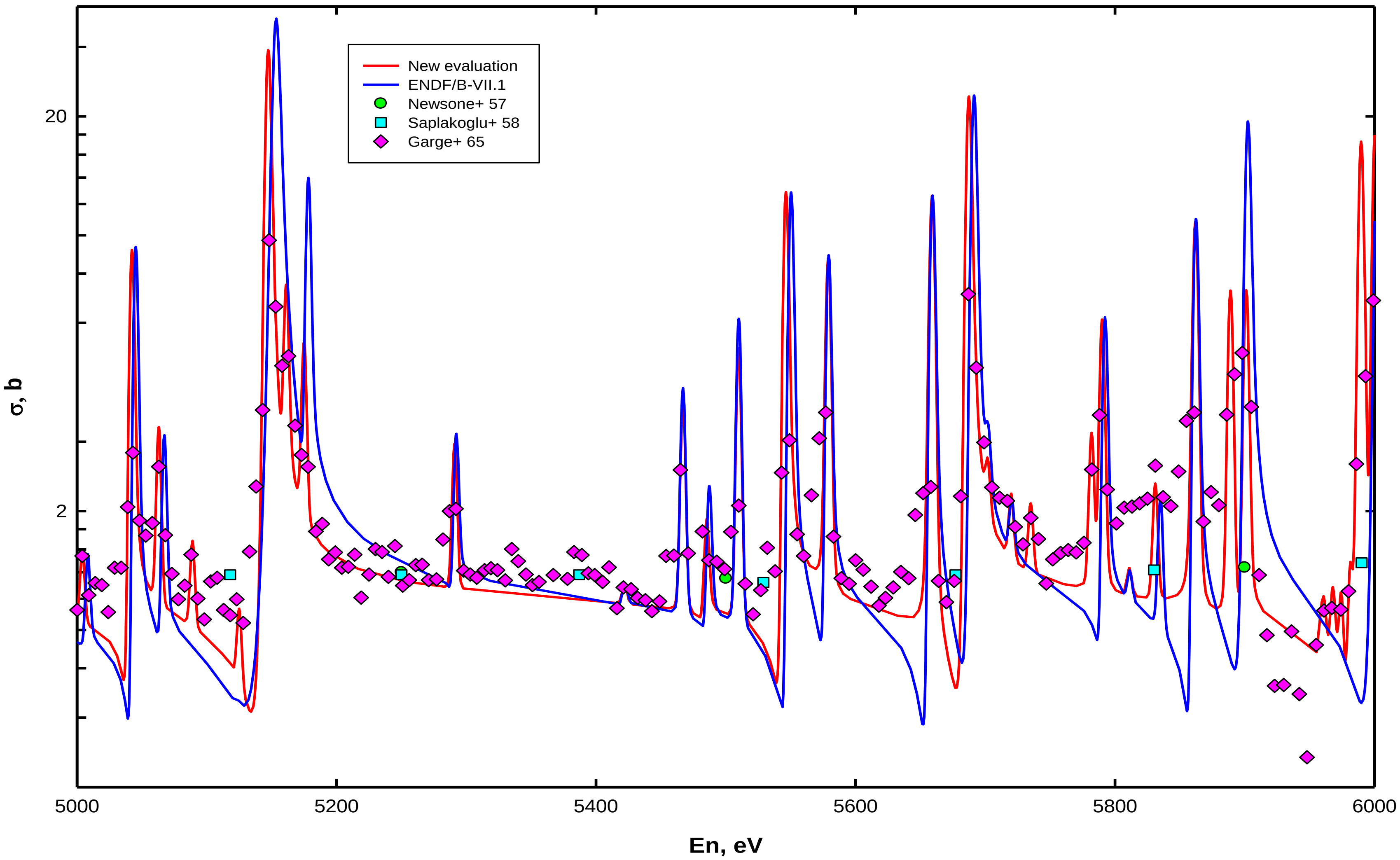

FIG. 7.3. Reconstructed from the evaluated resonance parameters ^{93}Nb(n,tot) cross sections in the energy range 5000-6000 eV in comparison with equivalent ENDB/B-VII.1 and experimental data.

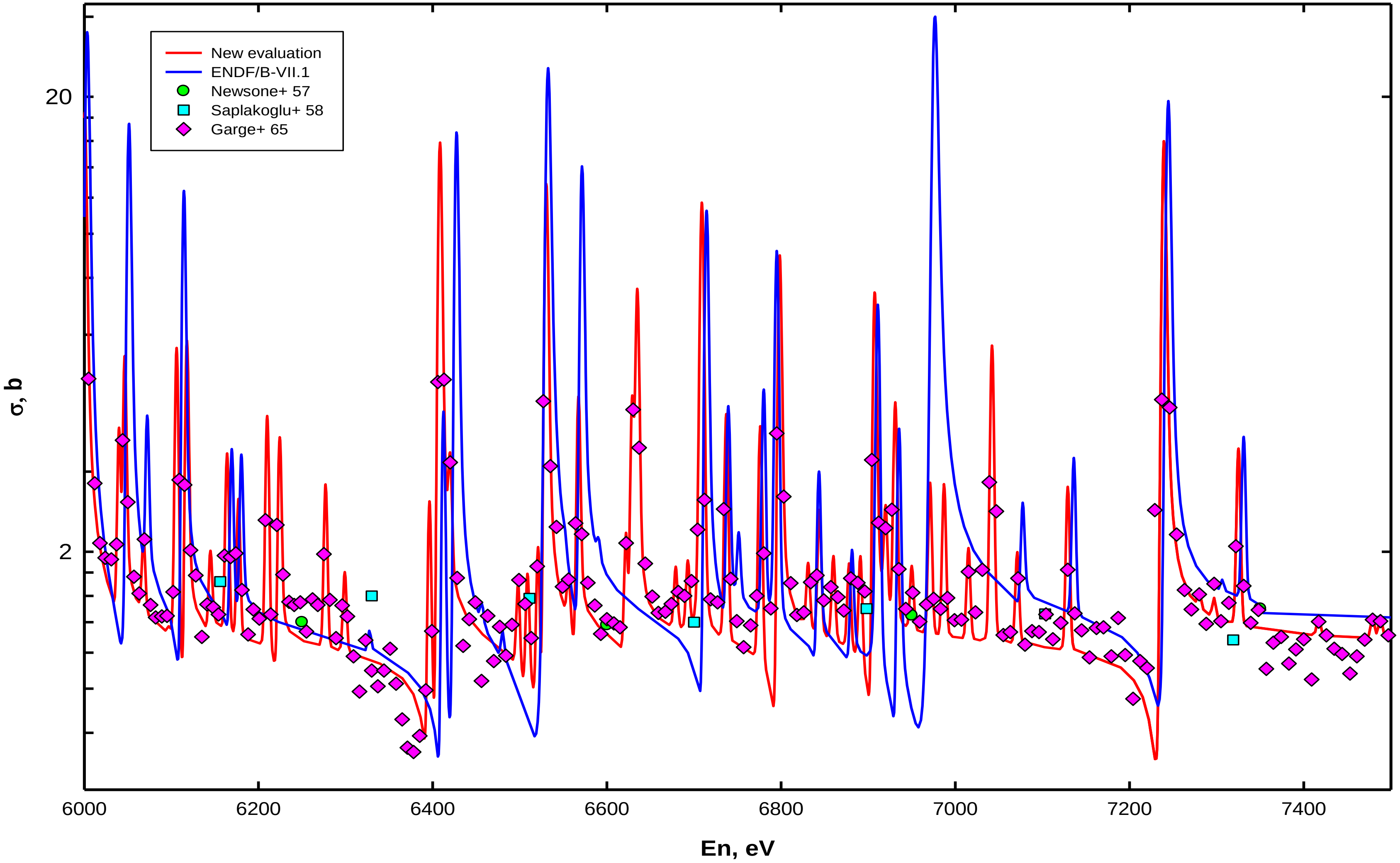

FIG. 7.4 Reconstructed from the evaluated resonance parameters ^{93}Nb(n,tot) cross sections in the energy range 6000-7500 eV in comparison with equivalent ENDB/B-VII.1 and experimental data.

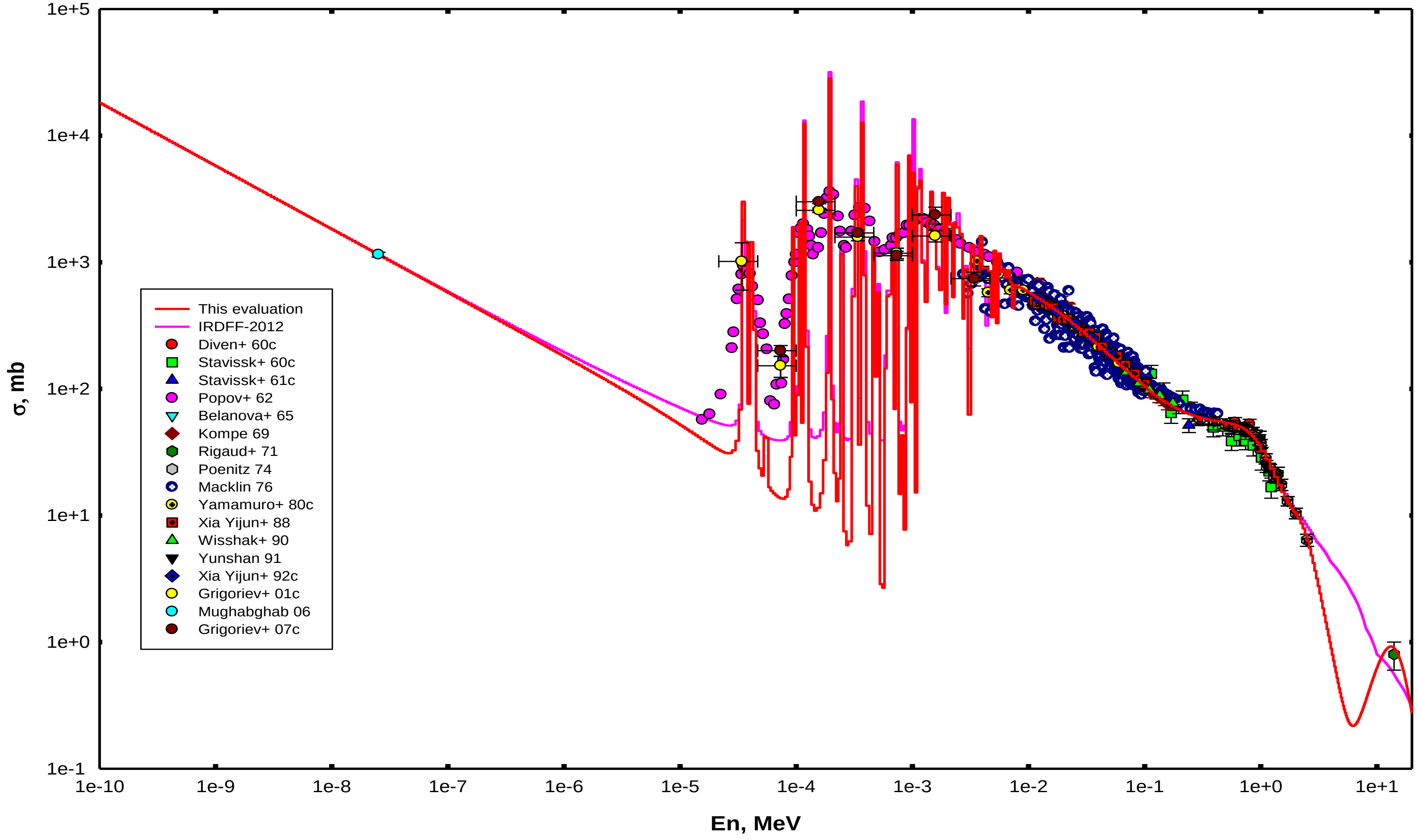

FIG. 7.5 Re-evaluated $^{93}Nb(n,\gamma)^{94}Nb$ reaction excitation function in the energy range 1.00E-10 eV – 20 MeV in comparison with equivalent IRDFF-2012 (ENDB/B-VII.1) and experimental data.

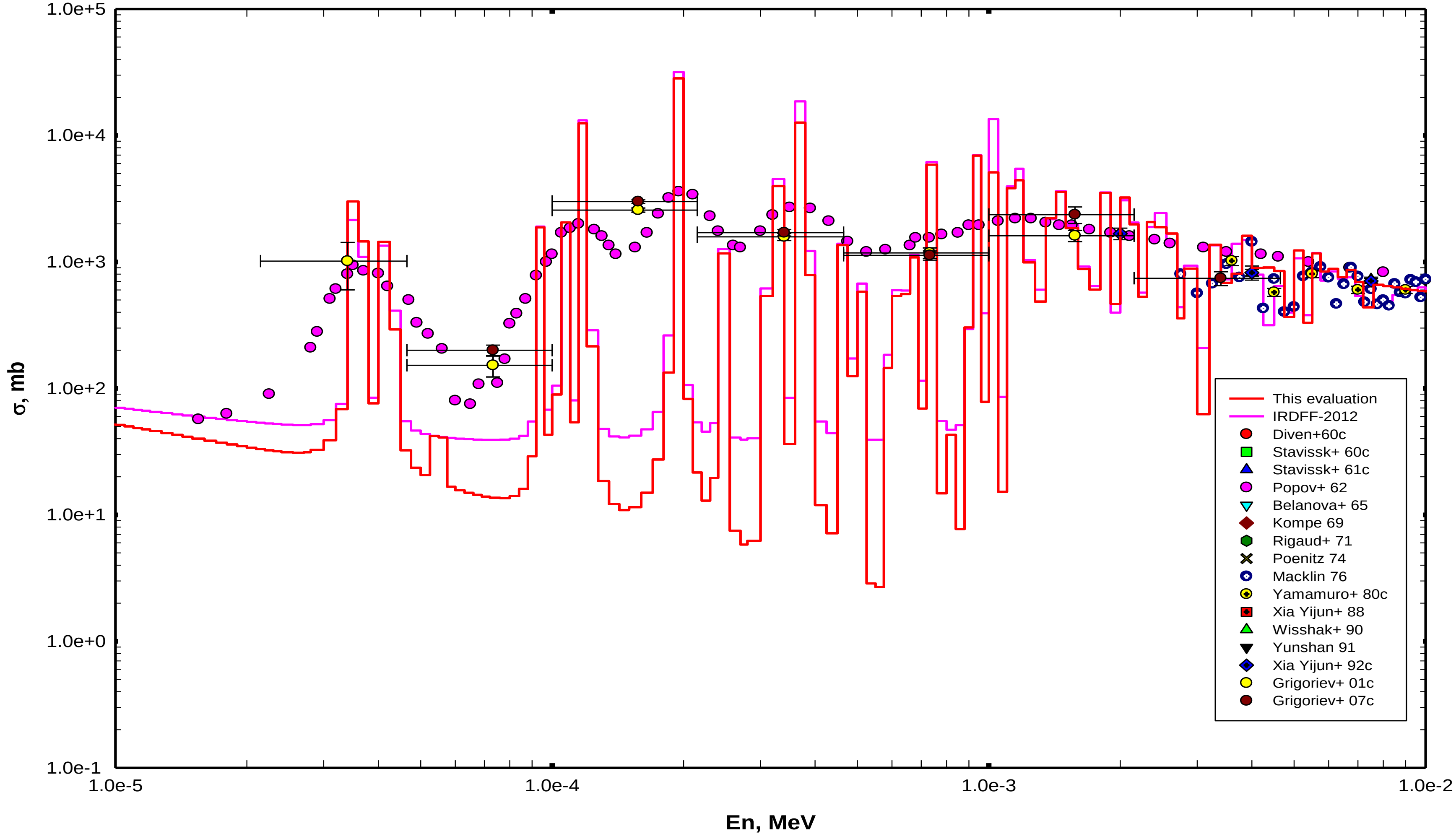

FIG. 7.6. Re-evaluated $^{93}Nb(n,\gamma)^{94}Nb$ reaction excitation function in the energy range 1.00E-5 eV – 1.00E-2 MeV in comparison with equivalent IRDFF-2012 (ENDB/B-VII.1) and experimental data.

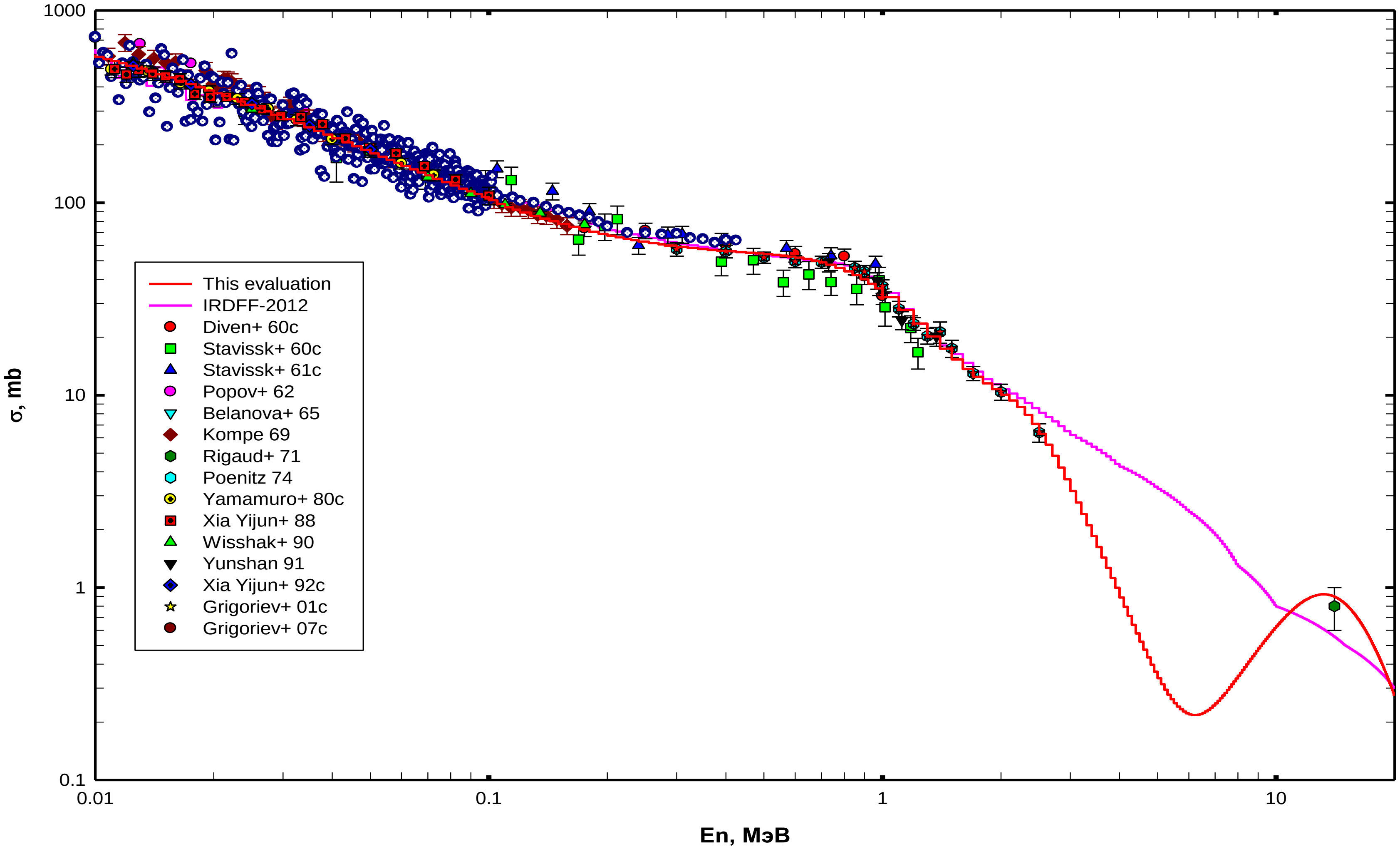

FIG. 7.7. Re-evaluated ^{93}Nb(n,γ)^{94}Nb reaction excitation function in the energy range 0.01 – 20 MeV in comparison with equivalent IRDFF-2012 (ENDB/B-VII.1) and experimental data.

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8. EVALUATION OF EXCITATION FUNCTION OF THE ^{113}In(n,n')^{113m}In REACTION

The isotopic abundance of ^{113}In in natural indium is (4.29 ± 0.05) atom percent. The 391.69-keV ($J^{\pi}=1/2^{-}$) metastable level of ^{113m}In excited in the (n,n') reaction undergoes 100% via IT decay with a half-life of (99.476 ± 0.023) minutes. The isomeric transition is accompanied with the emission of X-ray and gamma-ray radiation. The probability of emitted 391.70 keV gamma-rays per one decay is equal to I_{γ} = 0.6494 ± 0.0017. Recommended decay data for the half-life and gamma ray emission probability per decay of ^{113m}In were taken from Ref. [8.31].

Experimental information about the ^{113}In(n,n')^{113m}In reaction excitation function is given in the works [8.1-8.23] and covered neutron energies range from 448 keV to 17.83 MeV. Microscopic experimental data were analyzed during the preparation of the input database assembled in order to evaluate the cross sections and uncertainties for the ^{113}In(n,n')^{113m}In reaction.. During this procedure, data of works [8.1], [8.3-8.22] were corrected on the basis of the newly recommended cross section data for the relevant monitor reactions and the recommended decay data (see Table 2.1).

Excitation function of the ^{113}In(n,n')^{113m}In reaction in the energy range from threshold to 20 MeV was evaluated by means of statistical analyses of the experimental cross section data [8.1], [8.3], [8.5-8.12], [8.14-8.17], [8.19-8.23] and data from theoretical model calculations, which were used as an additional source of information between 15 - 20 MeV. Experimental data of Ai and Chou [8.8] at 1.41 MeV, 1.82 MeV, 2.02 MeV and 2.22 MeV were not taken into account in the evaluation due to significant underestimation of the ^{113}In(n,n')^{113m}In cross sections in these points.

Cross sections given in Refs. [8.2], [8.4], [8.11], [8.13] and [8.18] have been rejected completely due to their significant deviation from the main bulk of the experimental data. Within these rejected experimental data, the cross section values reported in Refs. [8.2], [8.11], [8.18] comprised only one energy point 14.7 MeV.

Cross section data and covariance matrix uncertainties for the ^{113}In(n,n')^{113m}In reaction were calculated simultaneously by means of the PADE-2 code. Uncertainties in the evaluated excitation function of the ^{113}In(n,n')^{113m}In reaction are given in the form of a relative covariance matrix for 49-neutron energy groups (LB = 5). Eigenvalues of the relative covariance matrix presented in File-33 were calculated by means of PADE-2 and tested in additional by COVEIG code [3.73]. Six-digit eigenvalues of the relative covariance matrix given in File-33 are as follows:

```
3.46310E-07    3.78551E-07    3.85653E-07    4.23845E-07
4.37943E-07    4.67061E-07    4.99883E-07    5.22489E-07
5.43417E-07    5.80575E-07    6.70050E-07    7.40351E-07
8.76412E-07    9.62510E-07    1.19831E-06    1.38493E-06
1.67390E-06    2.16340E-06    2.47981E-06    3.48060E-06
4.09340E-06    5.51503E-06    7.83425E-06    1.04888E-05
1.34799E-05    1.77240E-05    2.17511E-05    2.45900E-05
2.04907E-04    3.62052E-04    4.59262E-04    4.96702E-04
5.39416E-04    5.92045E-04    8.55242E-04    1.04515E-03
1.25438E-03    1.41922E-03    1.45239E-03    1.80420E-03
2.66266E-03    3.28049E-03    4.13535E-03    4.41864E-03
5.09262E-03    6.91240E-03    8.67907E-03    2.20170E-02
4.76253E-02
```

All of these eigenvalues are positive.

Evaluated group cross sections and their uncertainties for the $^{113}In(n,n')^{113m}In$ reaction excitation function are listed in Table 8.1. Group boundaries are the same as in File-33.

TABLE 8.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{113}In(n,n')^{113m}In$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 20 MeV.

Neutron energy (MeV) from to	**Cross section (mb)**	**Uncer-tainty (%)**	**Neutron energy (MeV) from to**	**Cross section (mb)**	**Uncer-tainty (%)**
0.395 - 0.500	0.885	21.79	5.500 - 6.000	297.360	2.69
0.500 - 0.600	2.579	7.28	6.000 - 6.500	286.442	3.04
0.600 - 0.700	5.794	6.19	6.500 - 7.000	272.962	3.04
0.700 - 0.800	15.563	5.92	7.000 - 7.500	262.502	2.95
0.800 - 0.900	28.367	5.66	7.500 - 8.000	255.629	2.96
0.900 - 1.000	35.781	4.56	8.000 - 8.500	251.147	3.00
1.000 - 1.100	45.558	3.12	8.500 - 9.000	247.058	2.93
1.100 - 1.200	65.687	3.99	9.000 - 9.500	240.751	2.84
1.200 - 1.300	85.216	3.70	9.500 - 10.000	229.421	3.20
1.300 - 1.400	102.854	3.14	10.000 - 10.500	211.198	4.13
1.400 - 1.500	120.527	2.68	10.500 - 11.000	186.624	5.04
1.500 - 1.600	138.914	2.36	11.000 - 11.500	158.750	5.33
1.600 - 1.700	158.076	2.13	11.500 - 12.000	131.584	4.84
1.700 - 1.800	177.795	1.94	12.000 - 12.500	108.056	3.80
1.800 - 1.900	197.693	1.79	12.500 - 13.000	89.284	2.60
1.900 - 2.000	217.289	1.69	13.000 - 13.500	75.047	1.62
2.000 - 2.250	248.869	1.58	13.500 - 14.000	64.535	1.09
2.250 - 2.500	284.657	1.54	14.000 - 14.500	56.858	1.01
2.500 - 2.750	305.155	1.50	14.500 - 15.000	51.252	1.20
2.750 - 3.000	310.605	1.45	15.000 - 16.000	45.608	1.92
3.000 - 3.500	299.094	1.64	16.000 - 17.000	40.920	3.57
3.500 - 4.000	274.110	2.36	17.000 - 18.000	38.184	5.54
4.000 - 4.500	260.676	2.24	18.000 - 19.000	36.497	7.51
4.500 - 5.000	271.612	2.13	19.000 - 20.000	35.398	9.36
5.000 - 5.500	292.834	2.20			

Uncertainties in the evaluated $^{113}In(n,n')^{113m}In$ excitation function range from 1.01% to 21.79%. The smallest uncertainties in the evaluated cross sections 1.01-1.20% are observed in the neutron energy range from 13.5 to 15.0 MeV. The highest uncertainty equal to 21.79% is near threshold. Relatively large uncertainties in cross sections exceeding 5% are in the energy range 0.5 – 0.9 MeV and between 17 – 20 MeV.

Figs. 8.1 and 8.2 show the evaluated excitation function for the $^{113}In(n,n')^{113m}In$ reaction over the neutron energy range from threshold to 13.0 MeV and over the 12 – 20 MeV interval in comparison with TENDL-2010, EAF-2010 and JEFF-3.1/A libraries and corrected experimental data. Comparison of the evaluated excitation functions with experimental data in the whole neutron energy range from threshold to 20.0 MeV is shown in Fig. 8.3.

Behaviour of the evaluated $^{113}In(n,n')^{113m}In$ excitation function in the energy range from threshold to 20 MeV is different from excitation functions given in the TENDL-2010, EAF-2010 and JEFF-3.1/A libraries. The evaluated $^{113}In(n,n')^{113m}In$ excitation function in the energy range from threshold to 20 MeV better agrees with corrected experimental data. Cross section data given in EAF-2010 and JEFF-3.1/A libraries are equal up to 20 MeV and therefore on Figs. 8.1-8.3 are presented only the JEFF-3.1/A data. The EAF-2010 and JEFF-3.1/A data files very slightly differ in values of threshold energy for the ^{113}In(n,n')^{113m}In excitation function: 397.5199 keV (EAF-2010) and 397.5000 keV (JEFF-3.1/A).

Integral experimental data for the ^{113}In(n,n')^{113m}In reaction are given in Refs. [8.24-8.30]. Two experiments were carried out in neutron fields similar to the ^{235}U thermal fission neutron spectrum [8.24-8.25], while six experiments were performed in a ^{252}Cf spontaneous fission neutron spectrum [8.24], [8.26-8.30]. Experimental data obtained for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were corrected to the new recommended cross sections for monitor reactions and decay data.

Kobayashi and Kimura measured the ^{113}In(n,n')^{113m}In integral cross sections for the ^{235}U thermal fission neutron spectrum in the core center of reactor YAYOI. The measured value after correction to the new standards is equal to (164.20 ± 13.63) mb. Integral cross section given in Ref. [8.25] was determined in the neutron spectra generated at facility with 90%-enriched ^{235}U fission plate converter. The measured cross section after correction to the new standards is equal to (168.47 ± 11.02) mb. The average weighted cross section determined from these two works is equal to $\langle\sigma\rangle_{U\text{-}235}$ = (166.80 ± 8.56) mb.

Measured integral cross sections for the ^{252}Cf spontaneous fission neutron spectrum extend over a range from (158.9 ± 8.74) mb [8.28] to (172.6 ± 10.17) mb [8.24]. The average weighted cross section determined from experimental data [8.27-8.30] is equal to $\langle\sigma\rangle_{Cf\text{-}252}$ = (161.20 ± 3.29) mb.

Evaluated experimental data for the ^{235}U thermal fission neutron spectrum and ^{252}Cf spontaneous fission neutron spectrum were used in benchmark calculations. The results of tests with the evaluated excitation function for the ^{113}In(n,n')^{113m}In reaction are given in Table 8.2. Calculated averaged cross sections from this work are compared with the equivalent TENDL-2010, EAF-2010 and JEFF-3.1/A data.

TABLE 8.2. CALCULATED AND MEASURED AVERAGED CROSS SECTIONS FOR THE ^{113}In(n,n')^{113m}In REACTION IN ^{235}U THERMAL FISSION AND ^{252}CF SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	**Averaged cross section, mb**		**C/E**
	Calculated	**Measured**	
^{235}U thermal fission neutron spectrum	155.08 [A] 155.35 [B] 144.54 [C] 144.55 [D]	166.80 ± 8.56 [*]	0.92974 0.93135 0.86655 0.86661
^{252}Cf spontaneous fission neutron spectrum	158.14 [A] 158.00 [B] 147.94 [C] 147.94 [D]	161.20 ± 3.29 [**]	0.98102 0.98015 0.91774 0.91774

[A] - Present evaluation.
[B] - TENDL-2010.
[C] - EAF-2010.
[D] - JEFF-3.1/A.
[*] - Average-weighted value obtained from the experimental data [8.24] and [8.25].
[**] - Average-weighted value obtained from the experimental data [8.27-8.30].

Data given in the Table 8.2 show that the ^{113}In(n,n')^{113m}In integral cross sections for ^{235}U thermal fission neutron spectrum calculated from the present evaluation and TENDL-2010, EAF-2010, JEFF-3.1/A libraries are lower in comparison with experimental data. The $<\sigma>_{U\text{-}235}$ cross sections calculated from new evaluation and TENDL-2010 excitation functions are differing by 7% from equivalent experimental data. Discrepancies between experimental value and results of calculation carried out on the basis of EAF-2010 and JEFF-3.1/A libraries are equal to 13.3%.

The C/E values obtained for the ^{252}Cf spontaneous fission neutron spectrum show that the integral cross sections calculated from newly evaluated and TENDL-2010 excitation functions agree well with the experimental values for this benchmark spectrum. The equivalent data calculated from EAF-2010 and JEFF-3.1/A libraries have discrepancies of about 8.2% with integral experimental data.

Present evaluation of the ^{113}In(n,n')^{113m}In reaction excitation function in comparison with TENDL-2010 evaluation agree well simultaneously with corrected microscopic experimental data and integral experimental data for the ^{252}Cf spontaneous fission neutron spectrum. The 90%-response ranges for the ^{113}In(n,n')^{113m}In excitation function in the ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are very similar: 1.2 – 5.9 MeV and 1.3 – 6.2 MeV, respectively. This fact permits to conclude that $<\sigma>_{U\text{-}235}$ cross section obtained from the experimental data [8.24] and [8.25] is overestimated by ≈ 7%. New precise measurements in the ^{235}U thermal fission neutron spectrum are desirable for the ^{113}In(n,n')^{113m}In reaction.

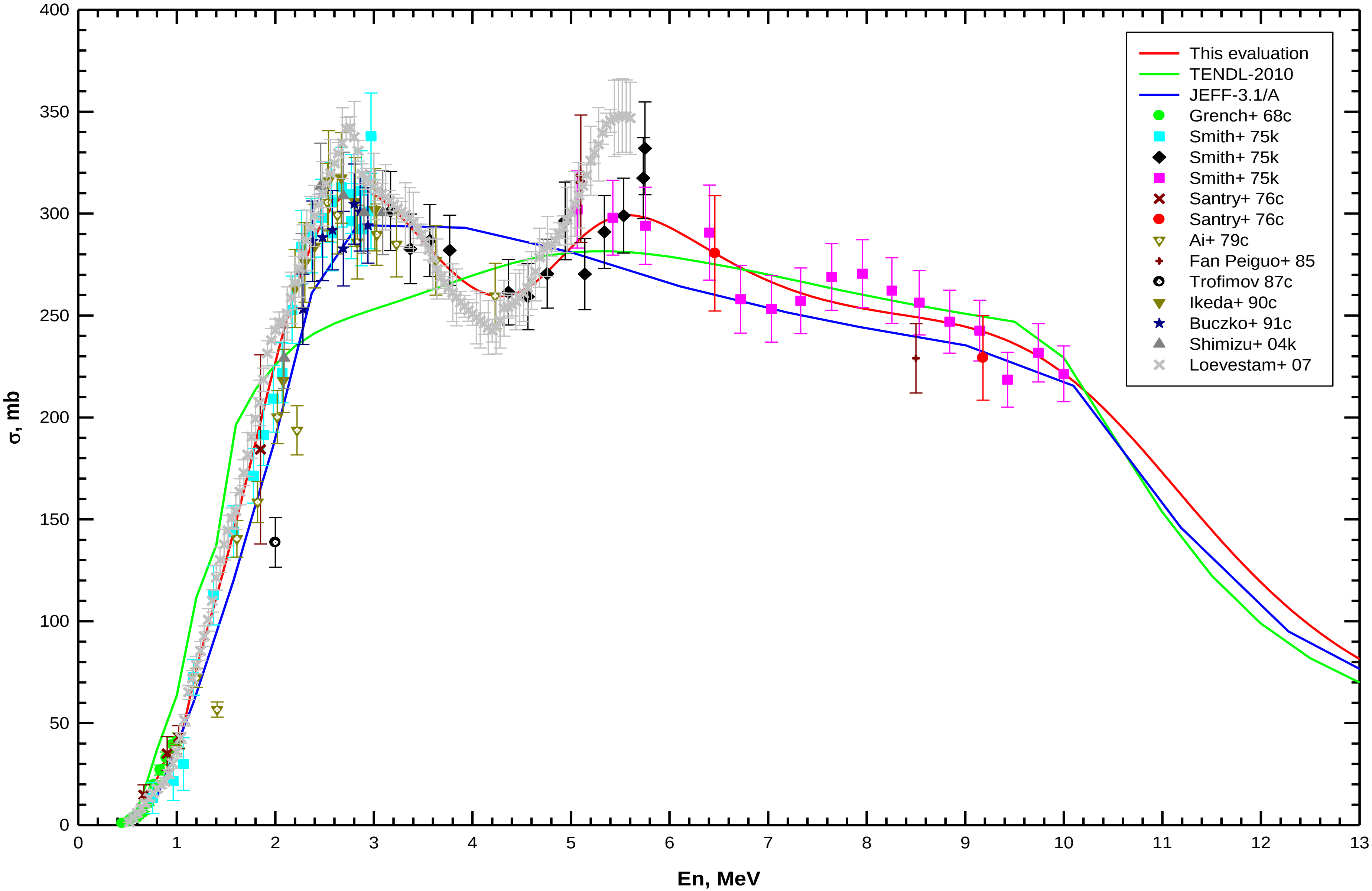

FIG. 8.1. Evaluated excitation function of the ^{113}In(n,n')^{113m}In reaction in the energy range from threshold to 13 MeV in comparison with TENDL-2010, JEFF-3.1/A and corrected experimental data.

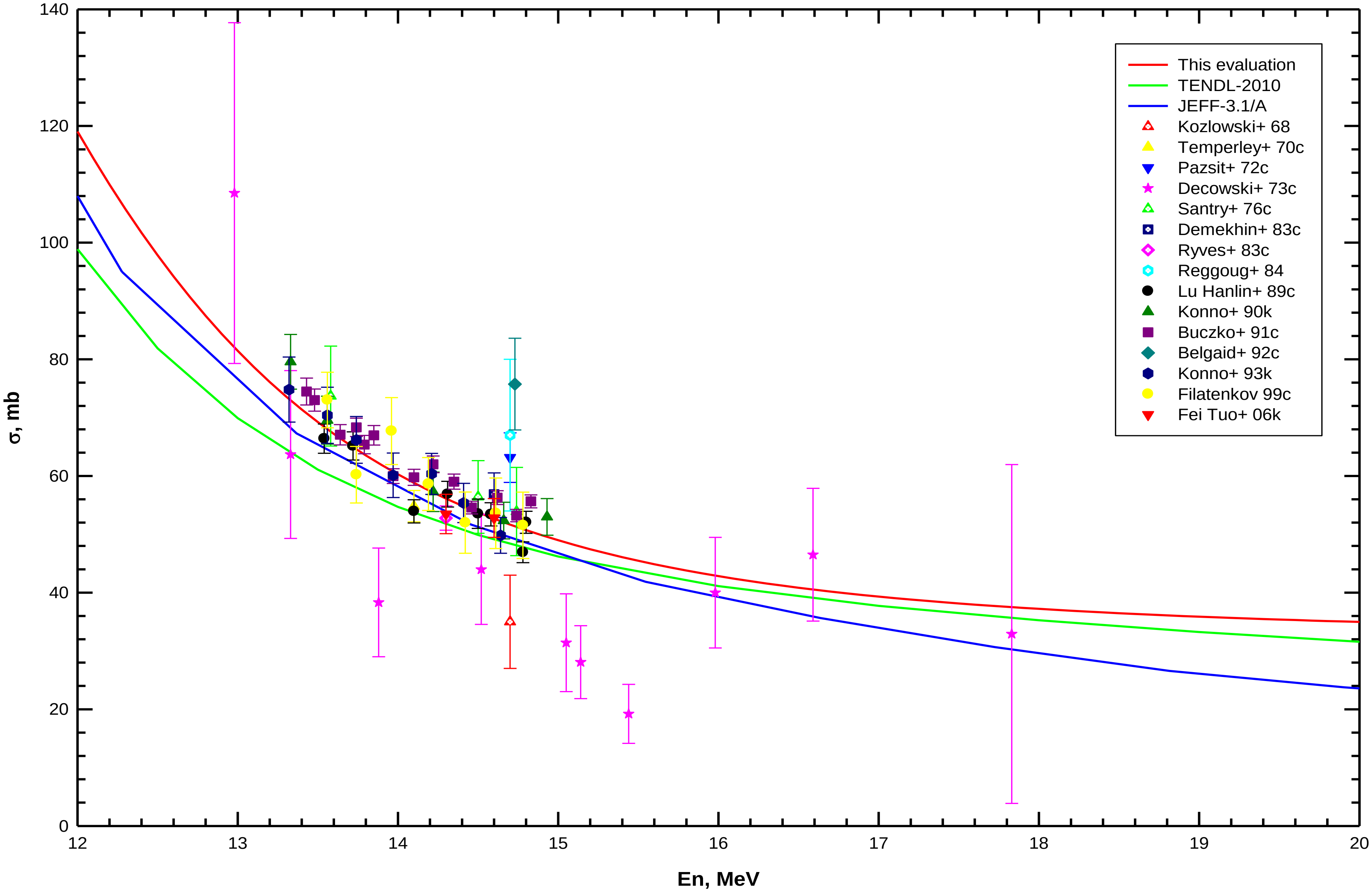

FIG. 8.2. Evaluated excitation function of the ^{113}In(n,n')^{113m}In reaction in the energy range from 12 to 20 MeV in comparison with TENDL-2010, JEFF-3.1/A and corrected experimental data.

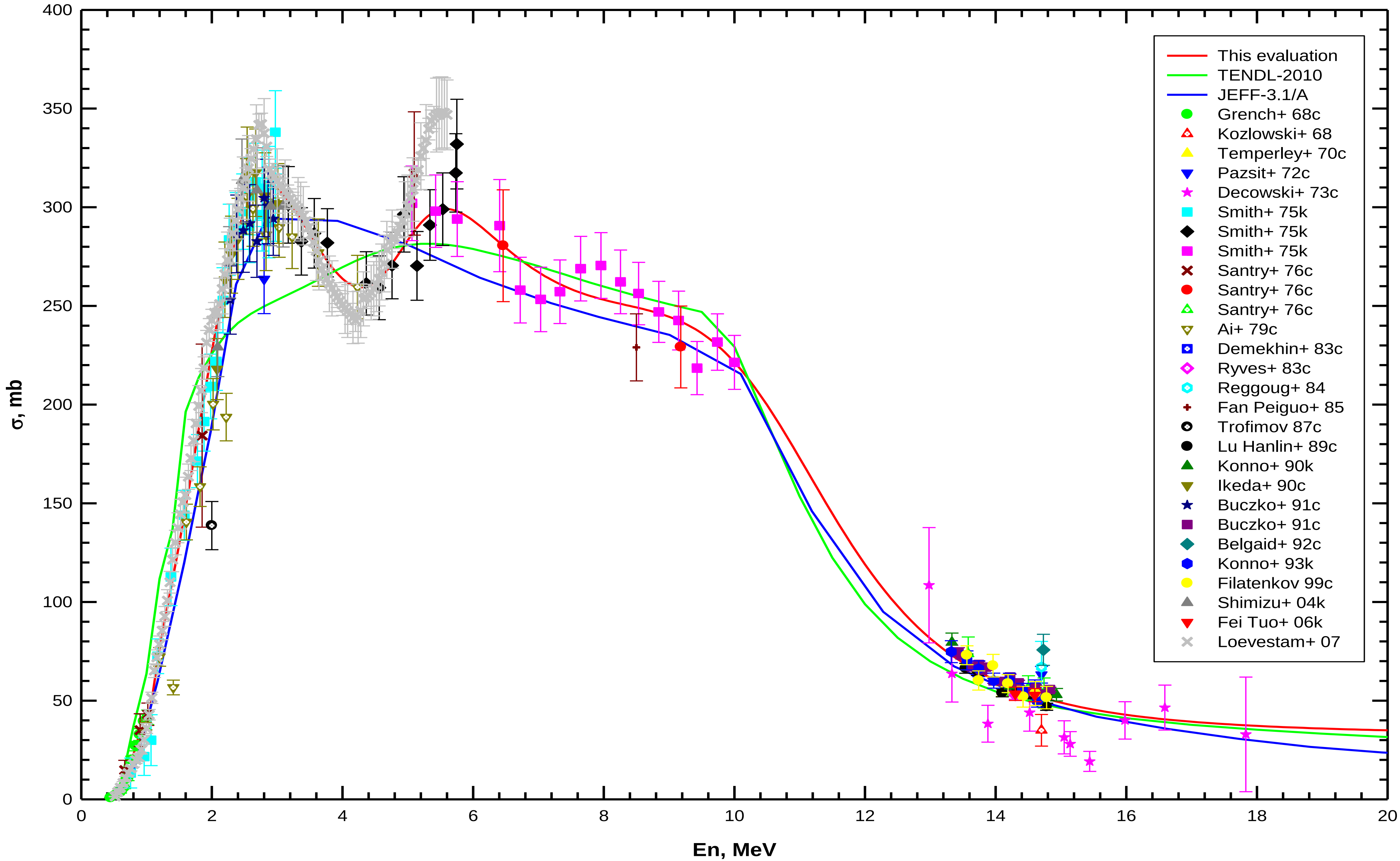

FIG. 8.3. Evaluated excitation function of the ^{113}In(n,n')^{113m}In reaction in the energy range from threshold to 20 MeV in comparison with TENDL-2010, JEFF-3.1/A and corrected experimental data.

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9. EVALUATION OF EXCITATION FUNCTION OF THE $^{115}In(n,\gamma)^{116m}In$ REACTION

The abundance of ^{115}In isotope in natural indium is (95.71 ± 0.05) atom percent. The first 127.267-keV ($J^{\pi} = 5^{+}$) metastable level of ^{116}In – $^{116m1}In$ exited in the (n,γ) reaction undergoes 100% via β^{-} decay with a half-life of (54.29 ± 0.17) minutes. The β^{-} transition is accompanied by emission of X-ray and gamma-ray radiation. The most intense lines in gamma-ray spectrum are 1097.23-keV line (I_{γ} = 0.585 ± 0.008) and 1293.56-keV line (I_{γ} = 0.848 ± 0.012). The second 289.660-keV ($J^{\pi} = 8^{-}$) metastable level of ^{116}In – $^{116m2}In$ excited in the (n,γ) reaction undergoes 100% via IT decay with a half-life of (2.18 ± 0.04) seconds to the first 127.267-keV ($J^{\pi} = 5^{+}$) metastable level. Recommended decay data for the half-life and radiation emission probabilities per decay of ^{116g}In, $^{116m1}In$ and $^{116m2}In$ were taken from Ref. [9.60].

For the dosimetry application it is necessary to know the total cross section for production of $^{116m1}In$ and $^{116m2}In$ in $^{115}In(n,\gamma)$ reaction. This combined reaction may be writen as $^{115}In(n,\gamma)^{116m}In$.

Microscopic experimental data [9.1-9.59], [9.61-9.70] were analyzed in order to evaluate radiative capture cross section for the ^{115}In, isomeric ratios $R_g(E) = \sigma_g(E)/\sigma_{m+g}(E)$, $R_m(E) = \sigma_m(E)/\sigma_{m+g}(E)$ and the $^{115}In(n,\gamma)^{116g}In$ and $^{115}In(n,\gamma)^{116m}In$ partial reaction excitation functions in a wide energy region 1.0E-05 eV – 20 MeV.

Experimental data of works [9.2], [9.4-9.6], [9.9-9.12], [9.14-9.17], [9.19-9.28], [9.30-9.31], [9.33-9.43], [9.45-9.48], [9.50-9.52], [9.54-9.55] and [9.57-9.59] in the process of analysis were corrected to the new standards for the relevant monitor reactions (see Table 2.1) and to the recommended decay data for ^{116g}In and ^{116m}In [9.60].

Data presented in the References [9.7], [9.11-9.13], [9.21], [9.33], [9.35], [9.54] and [9.57] were measured relative to the cross section of the $^{127}I(n,\gamma)^{128}I$ monitor reaction. Cross section data for the $^{127}I(n,\gamma)^{128}I$ reaction used in the renormalization were obtained in framework of this RC. New re-evaluation of the $^{127}I(n,\gamma)^{128}I$ reaction excitation function was carried out in the energy range 3 keV – 20 MeV.

Special correction was done for experimental data [9.46] and [9.49]. After correction to the new standards, experimental data by Chi-Fong Ai and Jen-Chang Chou [9.46] were renormalized to the integral of experimental data of Cox [9.22] in overlapping energy interval 0.99 - 1.374 MeV. Original experimental data of H.A. Husain and S.E. Hunt [9.49] in the energy range 2.442 - 4.504 MeV were corrected to the preliminary evaluated cross section of 48.38 mb at 3 MeV, determined from the representative experimental data [9.48], [9.51], [9.55], [9.59]. Correction factors for the experimental data [9.46], [9.49] were equal to Fc = 0.87226 and Fc = 0.50660, respectively.

Measured by J. Bacso et al. [9.18] isomeric ratio $\sigma_g(E)/\sigma_m(E)$ at neutron energies 24 keV, 3.1 MeV, 14.76 MeV were used to determine absolute cross sections for the $^{115}In(n,\gamma)^{116g}In$ and $^{115}In(n,\gamma)^{116m+g}In$ reactions at these energy points. The $^{115}In(n,\gamma)^{116m}In$ reaction cross section was used as a reference data. Experimental data of J. Bacso et al. at 24 keV were obtained by means of Sb-Be source. The reference cross section at 24 keV point 759 mb (±14.6 %) was measured by R.L. Macklin et al. in the same way [9.6]. Experimental data of J. Bacso et al. at 3.1 MeV and 14.76 MeV were determined by means of $D(d,n)^{3}He$ and $T(d,n)^{4}He$ sources. The reference cross section at 3.1 MeV 42.66 mb (±4.0 %) was obtained from the representative experimental data [9.48], [9.51], [9.55], [9.59]. The reference cross section at 14.67 MeV 0.975 mb (±12.3 %) was evaluated from the representative experimental data [9.40], [9.47], [9.50]. The $^{115}In(n,\gamma)^{116g}In$ reaction cross sections obtained as described above from the experimental data of J. Bacso et al. [9.18] at 24 keV, 3.1 MeV, 14.76 MeV are equal to 210.24 mb (±14.6 %), 8.57 mb (±10.8 %) and 0.189 mb (±19.4 %), respectively. The $^{115}In(n,\gamma)^{116m+g}In$ reaction cross sections at 24 keV, 3.1 MeV, 14.76 MeV were determined by summing up as described above the $^{115}In(n,\gamma)^{116g}In$ data with the $^{115}In(n,\gamma)^{116m}In$ data used as a reference cross sections. The ^{115}In total capture cross sections at 24 keV, 3.1 MeV, 14.76 MeV are equal to 969.24 mb (±14.7 %), 51.23 mb (±5.15 %) and 1.164 mb (±13.49 %), respectively.

All available experimental data for the ^{115}In neutron resolved resonances are given in the works [9.61-9.80]. Experimental information covers the neutron energy interval from 1.457 eV to 2003.7 eV. The most part of experimental data were obtained for lower-lying resonances. New evaluation of resonance parameters of ^{115}In was performed by S.F. Mughabghab on the basis experimental data presented in the works [9.61-9.79]. In the publication [2.21] S.F. Mughabghab gives information for 253 resonances. Parameters of 18 resonances: E_r – energy of resonance, L – neutron orbital moment, J^{π} – spin and parity of resonance, Γ_n – neutron width, Γ_γ – radiative width are completely identified. All the remaining resonances are described in the work [2.21] with incomplete sets of parameters.

The $^{115}In(n,el)$, $^{115}In(n,\gamma)$ and $^{115}In(n,tot)$ reaction cross sections in the energy range 1.0E-05 eV – 2.0 keV are given in the described new evaluation using Multi Level Breit-Wigner (MLBW) resonance parameters. The MLBW parameters for 253 resonances of 2 l-values recommended by S.F. Mughabghab were adopted in the new evaluation. The values of J^{π} and Γ_γ for 235 resonances absent in the Ref. [2.21] were adapted from the ENDF/B-VII.1 library.

The results of measurements of the $^{115}In(n,\gamma)^{116m}In$ reaction cross section for the thermal neutrons are presented in the works [9.2-9.3], [9.7-9.9], [9.16-9.17], [9.19-9.20], [9.30], [9.37], [9.42-9.43] and [9.58]. The more precise experimental data are given in publications [9.17], [9.30], [9.37]. The average weighted cross section determined from these three works is equal to $<\sigma>_{th}$ = (161.80 ± 1.36) b. The cross section recommended by S.F. Mughabghab at 0.0253 eV [2.21] is based on the original experimental data of W. Mannhart, σ_{th} = (162.3 ± 0.7) b [9.37].

Thermal cross section of the $^{115}In(n,\gamma)^{116g}In$ reaction had been measured by L. Seren et al. [9.3], P. Fettweis [9.16] and K.H. Beckurts et al. [9.17] using the activation method. In the first two works authors have taken for half-life of ^{116g}In a value $T_{1/2}$= 13 seconds and obtained a very similar results σ_{th} = (51.80 ± 5.18) b and σ_{th} = (52.0 ± 6.24) b, respectively. In the precise measurements carried out by K.H. Beckurts et al. the authors used reaction $^{197}Au(n,\gamma)^{198}Au$ for neutron flux monitoring. Thermal cross section for monitor reaction and half-life for ^{116g}In were taken equal to σ_M = (98.8 ± 0.3) b, $T_{1/2}$ = (14.12 ± 0.03) seconds. New recommended values for these data are equal to σ_M = (98.65 ± 0. 09) b [2.21], $T_{1/2}$ = (14.10 ± 0.03) seconds [9.60]. After correction to the new standards thermal cross section for the $^{115}In(n,\gamma)^{116g}In$ reaction obtained from experimental data of K.H. Beckurts et al. is equal to σ_{th} = (41.94 ± 0.99) b. The equivalent cross section evaluated by S.F. Mughabghab is about of 5% lower, σ_{th} = (40.0 ± 2.0) b [2.21].

Total capture cross section of ^{115}In at the neutron energy 0.0253 eV determined from precise experimental data [9.17], [9.30], [9.37] is equal to $<\sigma>_{th}$ = (203.74 ± 2.35) b. The equivalent cross section evaluated by S.F. Mughabghab is slightly lower, σ_{th} = (202.0 ± 2.0) b [2.21].

The ratios $R_g(E) = \sigma_g(E)/\sigma_{m+g}(E)$ and $R_m(E) = \sigma_m(E)/\sigma_{m+g}(E)$ obtained from the representative experimental data at 0.0253 eV are equal to $R_g(E)$ = 0.206 ± 2.63%, $R_m(E)$ = 0.794 ± 1.44%. The equivalent data determined from the S.F. Mughabghab evaluation are equal to $R_g(E)$ = 0.198 ± 5.10%, $R_m(E)$ = 0.803 ± 1.08%.

The isomeric ratios $R_g(E) = \sigma_g(E)/\sigma_{m+g}(E)$ and $R_m(E) = \sigma_m(E)/\sigma_{m+g}(E)$ in the energy region 0.5 eV - 2 keV were evaluated from analysis of resonance integral data measured for the $^{115}In(n,\gamma)^{116g}In$, $^{115}In(n,\gamma)^{116m}In$ and $^{115}In(n,\gamma)^{116m+g}In$ reactions [9.8], [9.17], [9.30], [9.42], [9.61], [9.65-9.68], [9.70], [9.73], [9.78-9.80]. Full information for determination of the isomeric ratios $R_g(E)$ and $R_m(E)$ are presented in the works [9.17], [9.68], [9.70]. Obtained on the basis of these works isomeric ratios are equal to $R_g(E)$ = 0.210 ± 5.73%, $R_m(E)$ = 0.790 ±

3.95%. The equivalent parameters calculated from resonance integrals evaluated by S.F. Mughabghab [2.21] slightly differ, $R_g(E) = 0.197 \pm 5.52\%$, $R_m(E) = 0.803 \pm 4.84\%$.

The ratios $R_g(E) = \sigma_g(E)/\sigma_{m+g}(E)$ and $R_m(E) = \sigma_m(E)/\sigma_{m+g}(E)$ in the energy region 2 keV – 20 MeV were determined from the evaluated excitation functions of the ^{115}In(n,γ)^{116m}In and ^{115}In(n,γ)$^{116m+g}$In reactions. In the interval 2-100 keV both ratios are constant and equal to $R_g(E) = 0.210$, $R_m(E) = 0.790$. In the interval 0.1-20 MeV parameter $R_g(E)$ is decreasing from 0.210 to 0.158, simultaneously the $R_m(E)$ is increasing from 0.790 to 0.842. In the RRDFF-2012 (ENDF/B-VII.1) libraries is given only the isomeric ratio $R_m(E) = \sigma_m(E)/\sigma_{m+g}(E)$. The isomeric ratio in the energy region 1.0E-10 - 20 MeV is constant and equal to $R_m(E) = 0.790$.

Excitation function of the ^{115}In(n,γ)$^{116m+g}$In reaction in the energy region from 2 keV to 20 MeV was evaluated by means of statistical analyses of the experimental cross section data [9.18], [9.22], [9.27], [9.31], [9.32], [9.41], [9.44] and data from theoretical model calculation, which were the main source of information between 2 - 20 MeV.

Excitation function of the ^{115}In(n,γ)^{116m}In reaction in the energy range from 2 keV to 20 MeV was evaluated by means of statistical analyses of the experimental cross section data [9.2], [9.6], [9.12], [9.14], [9.22-9.28], [9.34], [9.36], [9.40], [9.41], [9.46-9.57], [9.59] and data from theoretical model calculations were used as an additional source of information between 8 - 20 MeV.

Some of experimental data for the ^{115}In(n,γ)^{116m}In reaction mentioned above were used only partially. The ^{115}In(n,γ)^{116m}In reaction cross sections measured by A.E. Johnsrud et al. [9.14] with D(d,n)^{3}He source in the region 2.65 – 6.25 MeV were not included in the data base for the evaluation due to incomplete correction for deposit of low energy neutrons. For the same reason there were not taken into account the results of measurements by H.O. Menlove et al. [9.24] obtained with D(d,n)^{3}He, α-Be, T(d,n)He4 sources and experimental data of Chi-Fong Ai and Jen-Chang Chou [9.46] measured in the energy range 3.43-4.23 MeV. The results of absolute measurements of Grady et al. [9.53] obtained by means of Na-D(2)C and Na-Be neutron sources at 0.265 and 0.964 MeV, respectively, were taken into account in the evaluation. At the same time the data of this work obtained in the measurements with Sb-Be and La-Be sources for the neutron energies 0.023 and 0.770 MeV were rejected due to a large discrepancy with the main bulk of experimental data. Experimental data by R.P. Gautam et al. [9.57] determined at 0.97 MeV and 1.17 MeV were rejected due to significant overestimation of the ^{115}In(n,γ)^{116m}In reaction cross section.

Uncertainties in the evaluated ^{115}In(n,γ)^{116m}In reaction excitation function are given via two independent matrices.

In the RRR range 1.0E-5 eV - 2 keV uncertainties are given in the form of a diagonal matrix of uncertainties for the 35 neutron energy intervals (LB = 1). Uncertainties in cross sections were calculated by means of DSIGNG code [9.81] from uncertainties in MLBW parameters.

In the energy range 2 keV - 20 MeV uncertainties are presented in the form of relative covariance matrix for the 49-neutron energy groups (LB = 5). The covariance matrix was generated with the PADE-2 code.

Eigenvalues of the 6-th digits relative covariance matrix given in File-40 were calculated by means of PADE-2 code and tested in additional by COVEIG code. It was found that all of the eigenvalues are positive.

Evaluated group cross sections and related uncertainties for the ^{115}In(n,γ)^{116m}In reaction excitation function are listed in Table 9.1. Group boundaries are the same as in File-40.

TABLE 9.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE ^{115}In(n,γ)^{116m}In REACTION IN THE NEUTRON ENERGY RANGE FROM 1.00E-5 eV TO 20 MeV.

Neutron energy (eV) from - to	Cross section (b)	Uncer-tainty (%)	Neutron energy (eV) from - to	Cross section (b)	Uncer-tainty (%)
1.000E-5 - 1.000E-4	3.75205E+3	2.32	2.000E+4 - 3.000E+4	6.97527E-1	5.12
1.000E-4 - 1.000E-3	1.18731E+3	2.32	3.000E+4 - 4.000E+4	5.86963E-1	4.58
1.000E-3 - 1.000E-2	3.77862E+2	2.32	4.000E+4 - 5.000E+4	5.09772E-1	4.08
1.000E-2 - 2.000E-2	2.08886E+2	2.32	5.000E+4 - 6.000E+4	4.54287E-1	3.69
2.000E-2 - 3.000E-2	1.62588E+2	2.32	6.000E+4 - 8.000E+4	3.97323E-1	3.28
3.000E-2 - 1.000E-1	1.10341E+2	2.33	8.000E+4 - 1.000E+5	3.42133E-1	2.91
1.000E-1 - 3.000E-1	7.60073E+1	2.33	1.000E+5 - 1.500E+5	2.81839- 1	2.57
3.000E-1 - 1.000E+0	1.16668E+2	2.35	1.500E+5 - 2.000E+5	2.32761E-1	2.32
1.000E+0 - 2.000E+0	3.39061E+3	3.30	2.000E+5 - 3.000E+5	1.97376E-1	2.18
2.000E+0 - 3.000E+0	4.22226E+1	2.29	3.000E+5 - 4.000E+5	1.74279E-1	2.24
3.000E+0 - 4.000E+0	1.36450E+2	5.70	4.000E+5 - 5.000E+5	1.65068E-1	2.42
4.000E+0 - 4.700E+0	1.86079E+1	3.75	5.000E+5 - 6.000E+5	1.61001E-1	2.64
4.700E+0 - 5.000E+0	4.09379E+0	3.45	6.000E+5 - 7.000E+5	1.61440E-1	2.86
5.000E+0 - 6.000E+0	2.45362E+0	6.17	7.000E+5 - 8.000E+5	1.62946E-1	3.07
6.000E+0 - 8.000E+0	2.24505E+0	21.43	8.000E+5 - 1.000E+6	1.67605E-1	3.37
8.000E+0 - 1.000E+1	1.46775E+2	48.83	1.000E+6 - 1.200E+6	1.74444E-1	3.82
1.000E+1 - 1.600E+1	3.06497E+0	38.17	1.200E+6 - 1.400E+6	1.77959E-1	4.29
1.600E+1 - 2.500E+1	8.20723E+0	6.31	1.400E+6 - 1.600E+6	1.72998E-1	4.65
2.500E+1 - 4.000E+1	1.02847E+1	6.64	1.600E+6 - 1.800E+6	1.59657E-1	4.79
4.000E+1 - 6.300E+1	1.89904E+0	11.27	1.800E+6 - 2.000E+6	1.40490E-1	4.69
6.300E+1 - 1.000E+2	5.78613E+0	8.44	2.000E+6 - 2.500E+6	1.04084E-1	4.29
1.000E+2 - 1.275E+2	1.73601E+0	15.42	2.500E+6 - 3.000E+6	6.22328E-2	4.01
1.275E+2 - 1.600E+2	2.80060E+0	18.79	3.000E+6 - 3.500E+6	3.74982E-2	4.08
1.600E+2 - 1.800E+2	9.02742E+0	10.42	3.500E+6 - 4.000E+6	2.36471E-2	4.29
1.800E+2 - 2.000E+2	7.32468E+0	12.27	4.000E+6 - 4.500E+6	1.56149E-2	4.65
2.000E+2 - 2.350E+2	9.17829E+0	24.54	4.500E+6 - 5.000E+6	1.07309E-2	5.24
2.350E+2 - 2.650E+2	5.14282E+0	12.31	5.000E+6 - 5.500E+6	7.57559E-3	6.11
2.650E+2 - 3.150E+2	6.23146E+0	23.51	5.500E+6 - 6.000E+6	5.48028E-3	7.26
3.150E+2 - 4.000E+2	2.58397E+0	20.24	6.000E+6 - 6.500E+6	4.01696E-3	8.65
4.000E+2 - 5.250E+2	4.55401E+0	22.42	6.500E+6 - 7.000E+6	2.97863E-3	10.25
5.250E+2 - 7.200E+2	2.19822E+0	21.67	7.000E+6 - 7.500E+6	2.22330E-3	12.01
7.200E+2 - 1.000E+3	2.49842E+0	22.00	7.500E+6 - 8.000E+6	1.66962E-3	13.94
1.000E+3 - 1.200E+3	1.47287E+0	20.94	8.000E+6 - 9.000E+6	1.11972E-3	17.01
1.200E+3 - 1.400E+3	1.59017E+0	22.73	9.000E+6 - 1.000E+7	7.18706E-4	22.06
1.400E+3 - 2.000E+3	1.64432E+0	21.35	1.000E+7 - 1.100E+7	6.20435E-4	24.18
2.000E+3 - 3.000E+3	2.21960E+0	5.48	1.100E+7 - 1.200E+7	7.31103E-4	22.40
3.000E+3 - 4.000E+3	1.80822E+0	4.98	1.200E+7 - 1.300E+7	8.97371E-4	19.43
4.000E+3 - 6.000E+3	1.49372E+0	4.92	1.300E+7 - 1.400E+7	9.77326E-4	15.02
6.000E+3 - 8.000E+3	1.26024E+0	4.92	1.400E+7 - 1.500E+7	9.24201E- 4	13.47
8.000E+3 - 1.000E+4	1.11570E+0	5.09	1.500E+7 - 1.600E+7	8.00809E- 4	14.92
1.000E+4 - 1.500E+4	9.63296E-1	5.33	1.600E+7 - 1.800E+7	5.95241E- 4	18.09
1.500E+4 - 2.000E+4	8.26429E-1	5.41	1.800E+7 - 2.000E+7	3.68882E-4	26.75

Uncertainties in the evaluated ^{115}In(n,γ)^{116m}In excitation function range from 2.18% to 48.83%. The smallest uncertainties, lower than 3%, are observed in the neutron energy ranges from 1.0E-5 – 1.0 eV, 2 – 3 eV and 80 - 700 keV. Uncertainties in cross sections exceeding 10% are in the energy ranges 6 -16 eV, 40-63 eV, 0.1 – 2.0 keV and 6.5 - 20 MeV.

The results of the new evaluation of the ^{115}In(n,γ)$^{116m+g}$In, ^{115}In(n,γ)^{116g}In and ^{115}In(n,γ)^{116m}In reaction excitation functions are shown in Figs 9.1-9.5. The compared evaluated excitation functions are given in the SAND-II 640-energy groups. Figs. 9.1, 9.2 show the evaluated excitation function for the $^{115}In(n,\gamma)^{116m+g}In$ reaction over the neutron energy ranges 1.000E-10 - 2.000 E-3 MeV and 0.001 - 20.0 MeV in comparison with RRDFF-2012 (ENDF/B-VII.1) and corrected experimental data. Below 0.002 MeV both compared evaluations are equivalent and agreed well with experimental data of Beckurts et al. [9.17] and Yoon et al. [9.78]. Yoon et al. had measured neutron capture cross section of natural indium in the energy interval 0.00309 eV – 23.720 keV. Fig. 9.1 shows experimental data of Yoon et al. obtained in the energy interval 0.00309 - 13.810 eV. Above 13.810 eV in the experimental data are very many resonances which belong to the $^{113}In(n,\gamma)$ reaction. In the energies interval 2 keV - 20 MeV the new evaluation of the total radiative capture cross section for ^{115}In is in better agreement with corrected experimental data than the RRDFF-2012 (ENDF/B-VII.1) library.

The evaluated excitation function for the $^{115}In(n,\gamma)^{116g}In$ reaction is presented in Fig. 9.3 over the neutron energy range 0.001 - 20.0 MeV in comparison with corrected experimental data. Experimental data of Cox for the $^{115}In(n,\gamma)^{116g}In$ reaction [9.22] were corrected to the new standards for the monitor reaction, but were not corrected to the new recommended half-life value for ^{116g}In - $T_{1/2}$ = (14.10 ± 0.03) seconds [9.60]. Cox used for half-life of ^{116g}In a value $T_{1/2}$ = 13 seconds. This value of half-life results in overestimation of the $^{115}In(n,\gamma)^{116g}In$ reaction cross section by about 30%. Grench and Menlove treated experimental data using half-life value for ^{116g}In - $T_{1/2}$= (14.12 ± 0.03) seconds [9.27]. Fig. 9.3 shows that the evaluated excitation function for the $^{115}In(n,\gamma)^{116g}In$ reaction and data of Grench and Menlove agree well. The evaluated excitation function agrees well also with cross sections obtained from J. Bacso et al. experimental data [9.18].

The evaluated excitation function for the dosimetry reaction $^{115}In(n,\gamma)^{116m}In$ is presented in Figs. 9.4-9.5 over the neutron energy ranges 1.000E-10 - 2.000 E-3 MeV and 0.001 - 20.0 MeV in comparison with RRDFF-2012 (ENDF/B-VII.1) and corrected experimental data. Below 0.002 MeV both compared evaluations are equivalent and agree well with experimental data of Beckurts et al. [9.17]. In the energies interval 2 keV - 20 MeV the new evaluation for the $^{115}In(n,\gamma)^{116m}In$ reaction cross section agrees better with experimental data than with the RRDFF-2012 (ENDF/B-VII.1) library.

Integral experiments for the $^{115}In(n,\gamma)^{116m}In$ reaction are described in the works [9.82-9.90]. Integral cross sections presented in publications [9.82-9.85] were measured in the neutron fields with spectra which are similar to ^{235}U thermal fission neutron spectrum. Integral cross sections measured in the field of ^{252}Cf spontaneous fission neutron spectrum are presented in the works [9.86-9.89]. Only one experiment was performed in the SIGMA-SIGMA spectrum [9.90].

Experimental data obtained for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were corrected to the new recommended cross sections for monitor reactions and decay data (see Ref. [2.22] and [2.23] in section 2). Data of H. Pauw [9.86], Z. Dezso and J. Csikai [9.87], W. Mannhart and W.G. Alberts [9.88] were corrected also to the new recommended decay data for ^{116m}In [9.60].

The results of measurements for ^{235}U thermal fission neutron spectrum presented in the works [9.82-9.83] and [9.85] cannot be qualified as representative experimental data. Integral cross section values for the $^{115}In(n,\gamma)^{116g}In$ and $^{115}In(n,\gamma)^{116m}In$ reactions are given in the works [9.82-9.83] without uncertainties. Facilities and generated neutron spectra in both publications are described very poorly. The integral cross section of 160 mb measured by J.D. Jenkins and F.B. Kam in a bare U-235 assembly at APFA-III-GODIVA [9.85] reactor is given without uncertainty information. The $^{115}In(n,\gamma)$ integral cross section was computed by dividing measured saturated activity by flux level obtained from a combination of 17 foil results (4.81E+10 n/cm^2/sec). The representative experiment for determining integral cross sections in the ^{235}U thermal fission neutron spectrum was carried out at MOL by A. Fabry [9.84]. Measurements were performed at especially created facility at the BR1 reactor. The horizontal graphite thermal column with a very high degree of thermalization was used as initial source of neutrons. The ^{235}U thermal fission neutron spectrum was generated by means of 90%-enriched ^{235}U fission plate converter (FPC). Integral cross section for the ^{115}In(n,γ)^{116m}In reaction measured by A. Fabry and corrected to the new standards is equal to $<\sigma>_{U-235}$ = (124.50 ± 5.29) mb.

Integral cross sections for the ^{115}In(n,γ)^{116m}In reaction measured in the field of ^{252}Cf spontaneous fission neutron spectrum have a range of obtained results from (115.66 ± 8.04) mb [9.89] to (123.93 ± 53.25) mb [9.87]. Data of H. Pauw (123.67 ± 4.25) mb [9.86], Z. Dezso and J. Csikai (123.93 ± 53.25) mb [9.87], W. Mannhart and W.G. Alberts (122.56 ± 3.25) mb) [9.88] agree excellently between each other. Average-weighted value obtained from these experimental data is equal $<\sigma>_{Cf-252}$ = (123.00 ± 2.58) mb.

Evaluated excitation function for the ^{115}In(n,γ)^{116m}In reaction was tested against the above mentioned integral experimental data. Calculated averaged cross sections over ^{235}U thermal fission neutron spectrum, ^{252}Cf spontaneous fission neutron spectrum and SIGMA-SIGMA spectrum are compared with the RRDFF-2012 (ENDF/B-VII.1) and experimental data in Table 9.2. Given in the last column is the ratio of the calculated to experimental cross sections (C/E).

TABLE 9.2. CALCULATED AND MEASURED AVERAGED CROSS SECTIONS FOR THE ^{115}In(n,γ)^{116m}In REACTION IN THE THREE NEUTRON SPECTRA.

Type of neutron field	**Averaged cross section, mb**		**C/E**
	Calculated	**Measured**	
^{235}U thermal fission neutron spectrum	126.33 [A] 126.17 [B]	124.50 ± 5.29 [9.84]	1.01470 1.01341
^{252}Cf spontaneous fission neutron spectrum	123.65 [A] 123.72 [B]	123.00 ± 2.58 [*]	1.00528 1.00585
SIGMA-SIGMA neutron spectrum	280.13 [A] 288.60 [B]	260.20 ± 9.00 [9.90]	1.07659 1.10915

[A] - Present evaluation.

[B] - RRDFF-2012 (ENDF/B-VII.1).

[*] - Average-weighted value evaluated in this work from corrected to the new standards integral experimental data [9.86-9.88].

The C/E values obtained for the ^{235}U thermal fission neutron spectrum and ^{252}Cf spontaneous fission neutron spectrum show that the ^{115}In(n,γ)^{116m}In integral cross sections calculated from newly evaluated and RRDFF-2012 (ENDF/B-VII.1) excitation functions agree well with the relevant experimental values. Discrepancies of 0.5-0.6% between calculated and experimental values $<\sigma>_{Cf-252}$ are low and do not exceed uncertainty of the experimental value. Discrepancies of 1.34-1.47% between calculated and experimental values $<\sigma>_{U-235}$ also do not exceed the uncertainty of the experimental value.

The integral cross section $<\sigma>_{\Sigma-\Sigma}$ in the SIGMA-SIGMA neutron field for the compared evaluations was calculated using neutron spectrum from a data file prepared by NDS of IAEA (see description in the section 3, page 15). The 90%-response range for the ^{115}In(n,γ)^{116m}In reaction in SIGMA-SIGMA neutron spectrum is lying between 36 keV - 1.40 MeV.

Data given in Table 9.2 show that calculated and measured integral cross sections $<\sigma>_{\Sigma-\Sigma}$ in SIGMA-SIGMA neutron field significantly differ: C/E = 1.07659 (for new evaluation) and C/E

= 1.10915 (for RRDFF-2012 (ENDF/B-VII.1) evaluation). Such a big difference between calculated and experimental values for $<\sigma>_{\Sigma\text{-}\Sigma}$ can be explained by two facts. Firstly, due to incomplete correction introduced to the experimental data for resonance self-shielding. Secondly, due to overestimation of neutron flux determined in the low-energy region part of the spectrum. The SIGMA-SIGMA neutron spectrum in the energy intervals 0.4 eV – 20 keV and 4 - 15 MeV was obtained by transport calculations performed using very old ENDF/B-III cross sections. As was mentioned in the Ref. [9.91] the neutron spectrum in the low-energy region is not so reliable due to the difficulties of preparing proper self-shielded cross sections for ^{238}U.

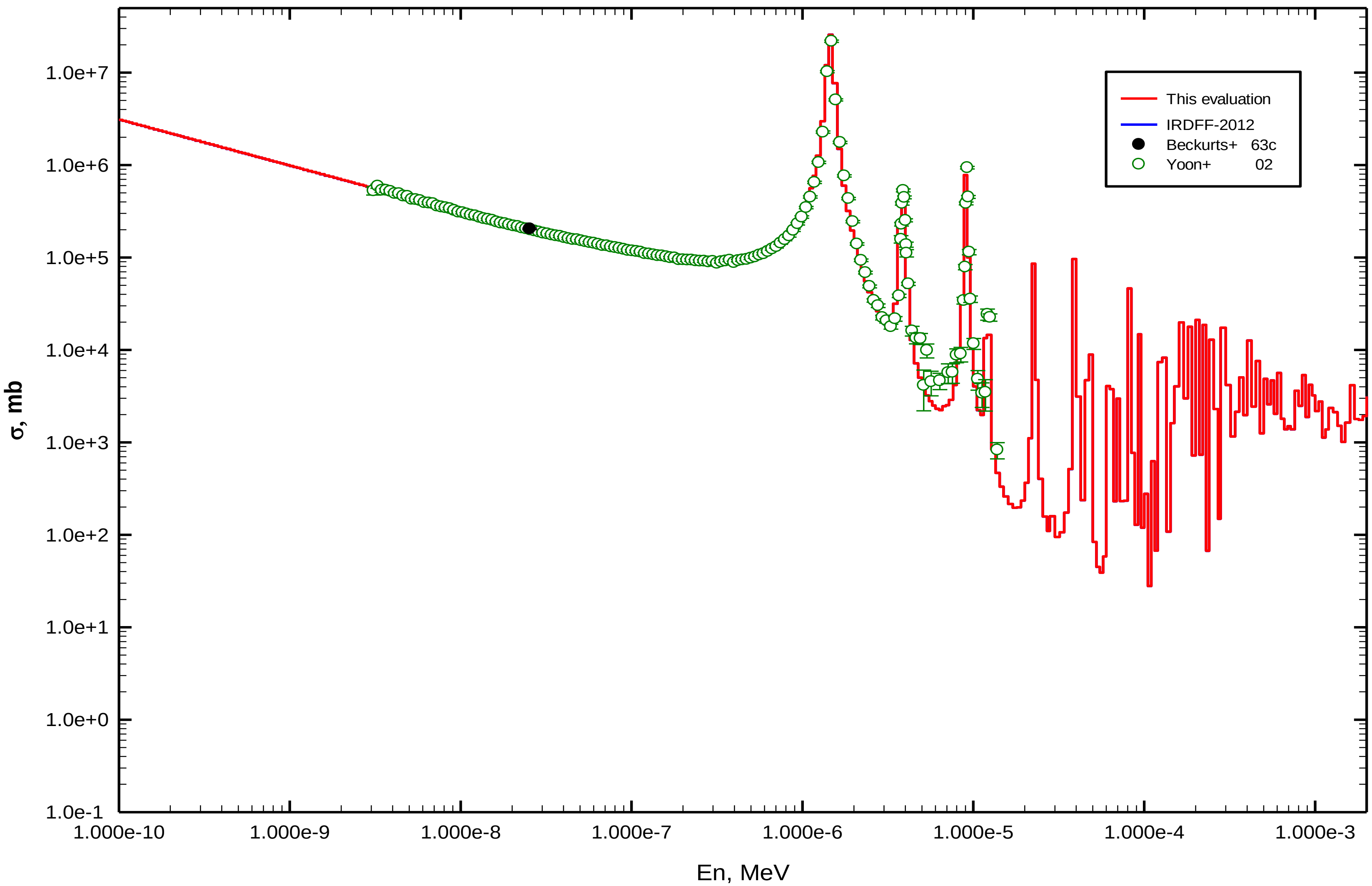

FIG. 9.1. Evaluated $^{115}In(n,\gamma)^{116m+g}In$ reaction excitation function at the energy range 1.000E-10 - 2.000E-3 MeV in comparison with IRDFF-2012 (ENDB/B-VII.1) and corrected experimental data.

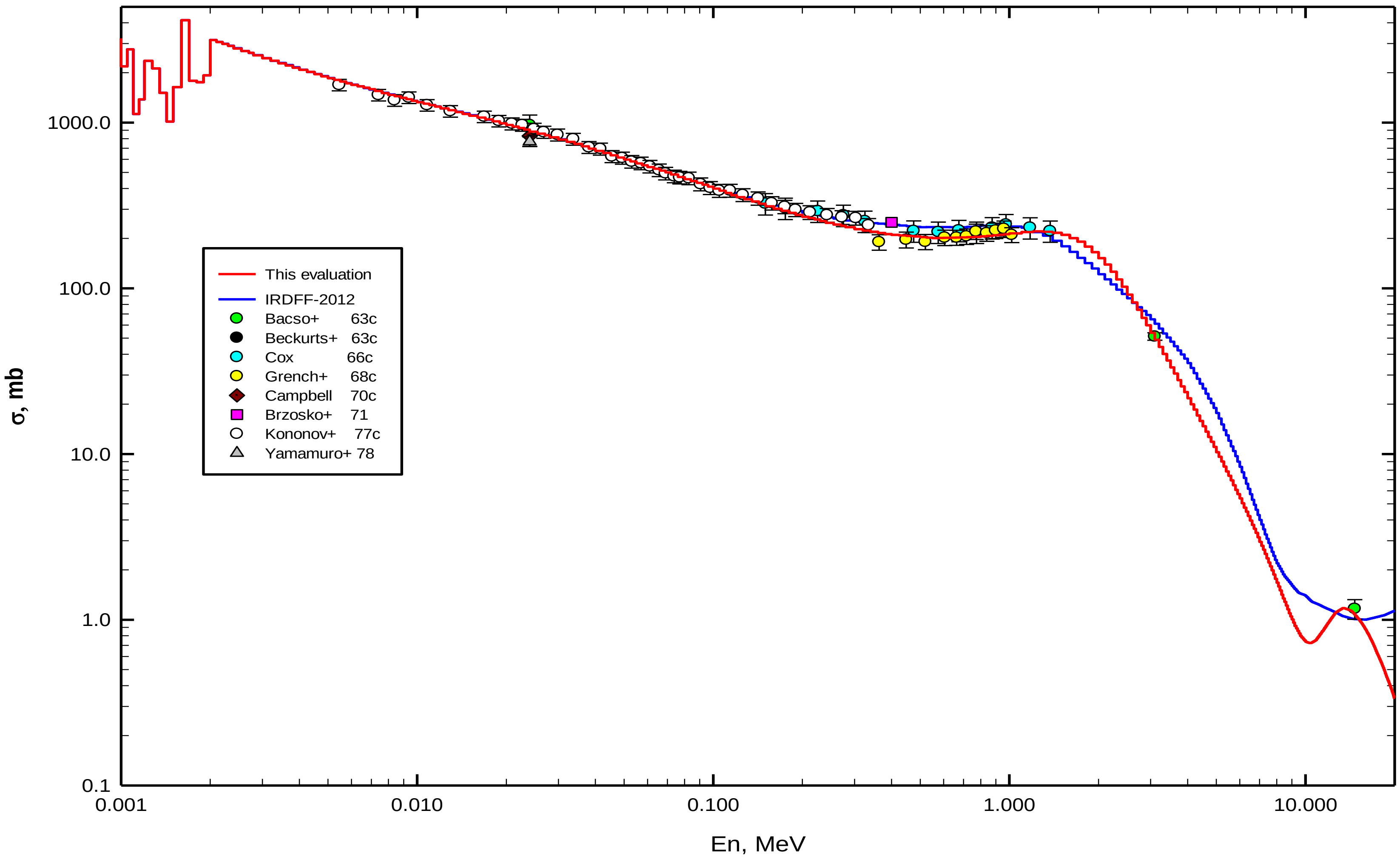

FIG. 9.2. Evaluated $^{115}In(n,\gamma)^{116m+g}In$ reaction excitation function at the energy range 0.001 - 20 MeV in comparison with IRDFF-2012 (ENDB/B-VII.1) and corrected experimental data.

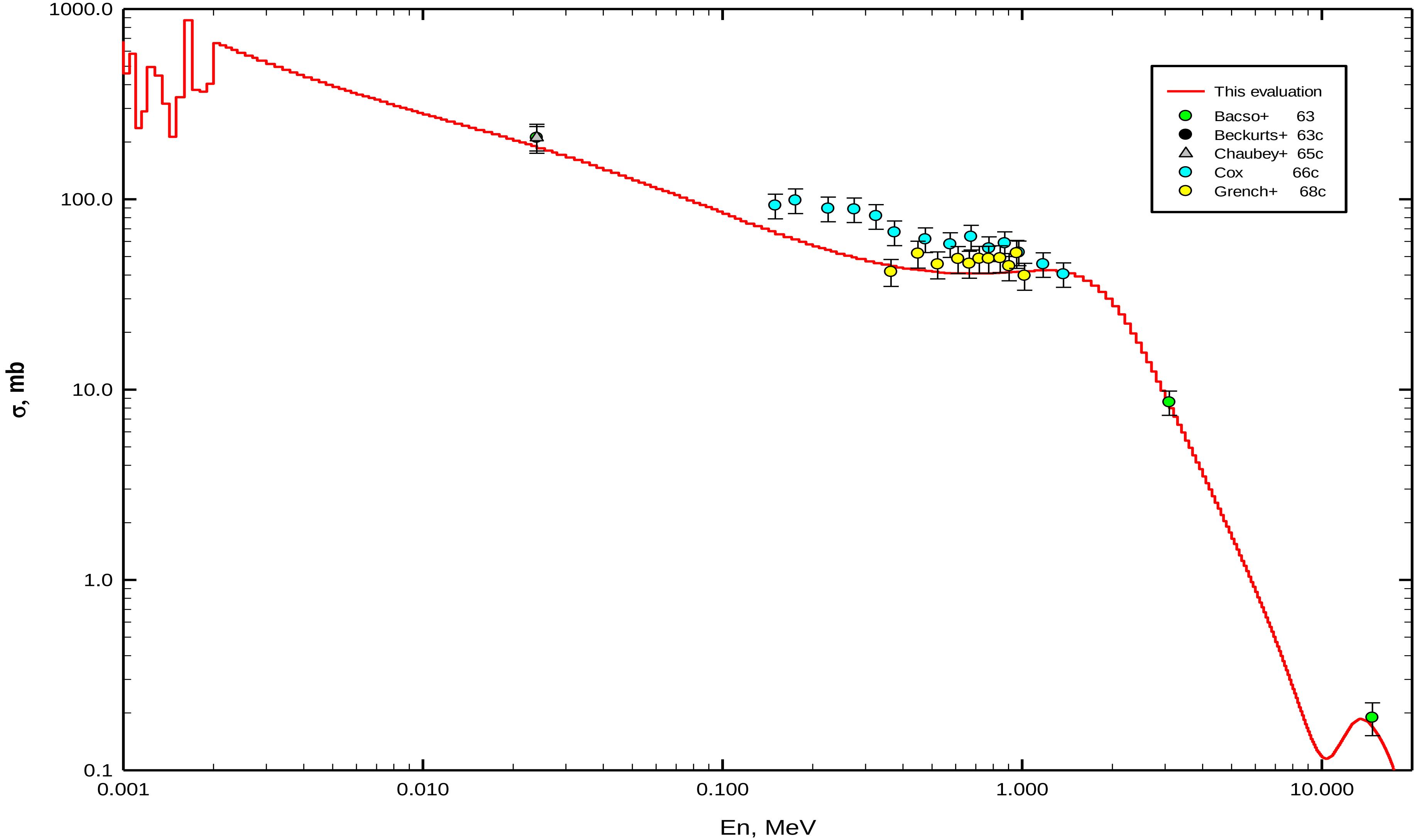

FIG. 9.3. Evaluated $^{115}In(n,\gamma)^{116g}In$ reaction excitation function at the energy range 0.001 - 20 MeV in comparison with corrected experimental data.

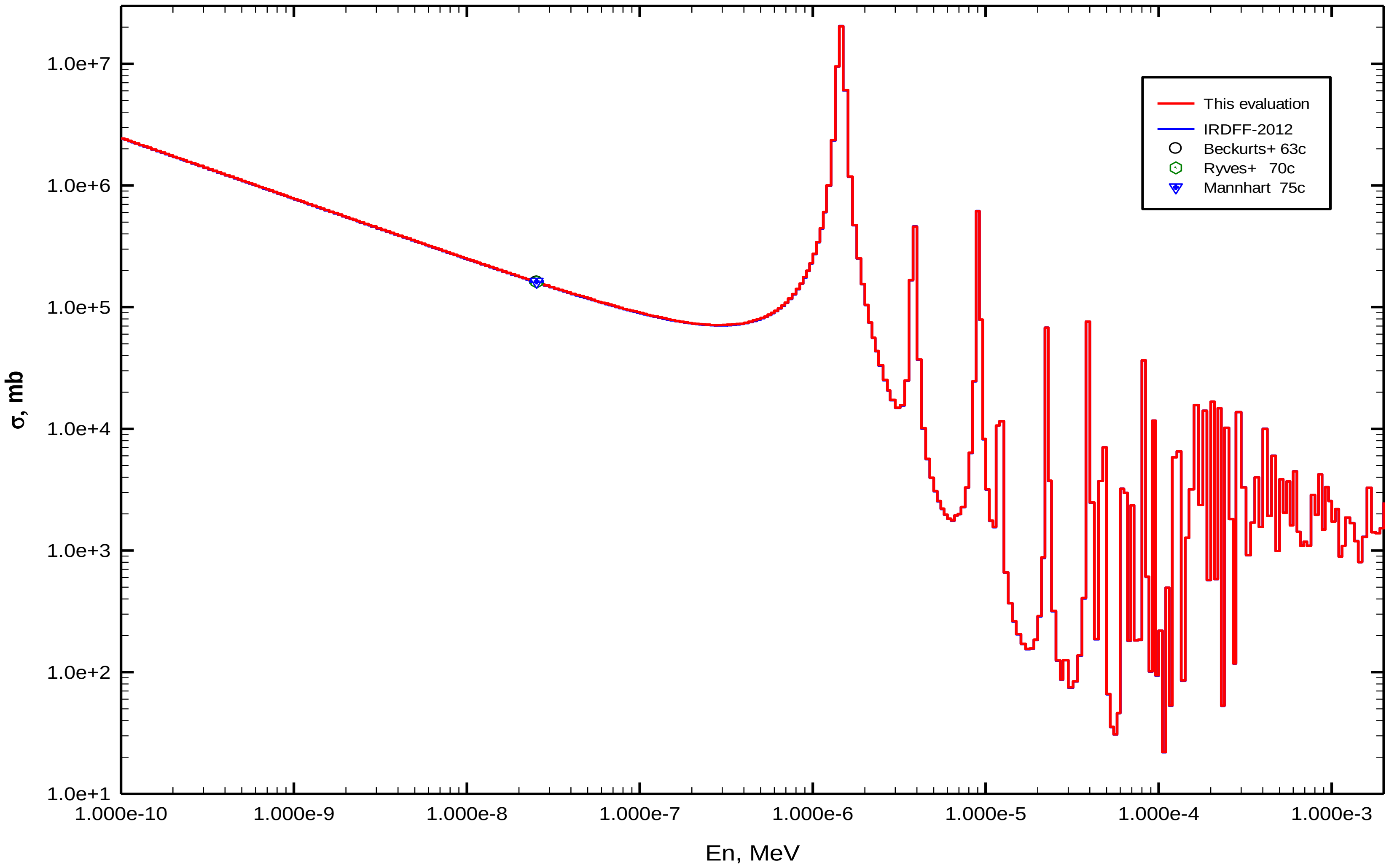

FIG. 9.4 Evaluated ^{115}*In(n,γ)*^{116m}*In reaction excitation function at the energy range 1.000E-10 - 2.000E-3 MeV in comparison with IRDFF-2012 (ENDB/B-VII.1) and corrected experimental data.*

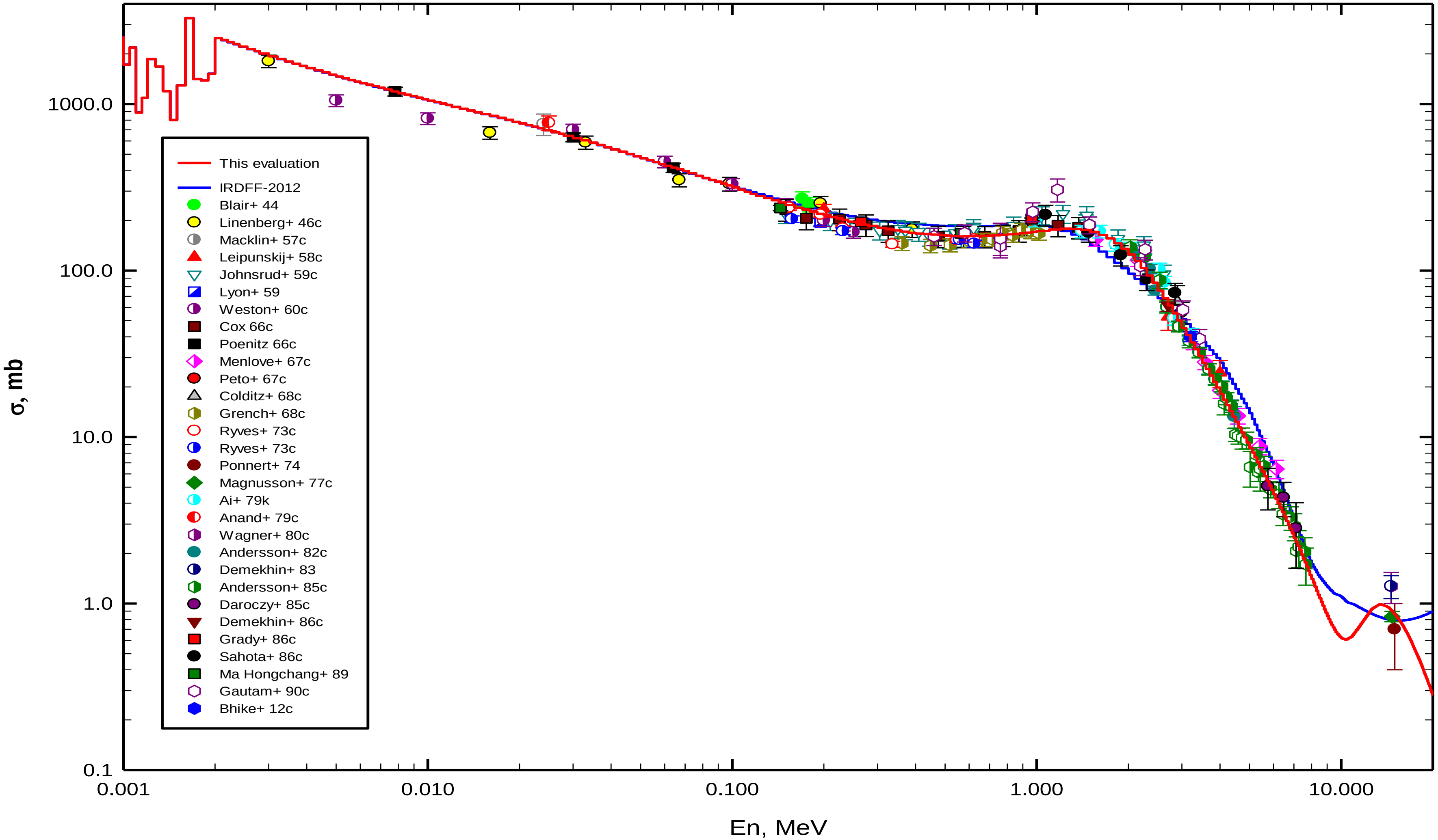

FIG. 9.5. Evaluated $^{115}In(n,\gamma)^{116m}In$ reaction excitation function at the energy range 0.001 - 20 MeV in comparison with IRDFF-2012 (ENDB/B-VII.1) and corrected experimental data.

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10. EVALUATION OF EXCITATION FUNCTION OF THE ^{169}Tm(n,3n)^{167}Tm REACTION

The isotopic abundance of ^{169}Tm in natural thulium is 100 atom percent. The ^{167}Tm produced by (n,3n) reaction undergoes 100 % via ε capture decay with a half-life of (9.25 ± 0.02) days. The ε decay is accompanied by emission of wide energy spread gamma-ray spectrum. The 207.8-keV gamma radiation (I_γ = 0.4102 ± 0.0777), is normally used to determine the ^{169}Tm(n,3n)^{167}Tm reaction rate. Recommended decay data for the half-life, and gamma-ray emission probabilities per decay of ^{167}Tm were taken from Ref. [2.11] of Section 2.
Experimental data for the ^{169}Tm(n,3n)^{167}Tm reaction excitation function are given only in four works [10.1-10.4] and cover neutron energies range from 15.09 MeV to 28.02 MeV. In the compilation [10.4] are presented revised experimental data from Ref. [10.3].

Microscopic experimental data were analyzed during the preparation of the input database assembled in order to evaluate the cross sections and uncertainties for the ^{169}Tm(n,3n)^{167}Tm reaction [10.1-10.2], [10.4]. During this procedure the experimental data given in Refs. [10.1] and [10.4] 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). Cross sections given in works [10.1] and [10.4] were measured by activation method and don't include uncertainty in the probability of 207.8-keV gamma radiation accompanying decay of ^{167}Tm. This uncertainty equal to 18.93% [2.11] was added to the activation experimental data.

The excitation function for the ^{169}Tm(n,3n)^{167}Tm reaction in the energy region from threshold to 60 MeV was evaluated by means of statistical analyses of the experimental cross section data [10.1], [10.2], [10.4] and data from theoretical model calculation carried out by means of the EMPIRE-2.19 code. Data from theoretical model calculation, which were the main source of information above 28 MeV, have been renormalized to the integral of the experimental data between 24 - 28 MeV.

Cross section data and covariance matrix uncertainties for the ^{169}Tm(n,3n)^{167}Tm reaction were calculated simultaneously by means of the PADE-2 code. Uncertainties in the evaluated excitation function of the ^{169}Tm(n,3n)^{167}Tm reaction are given in the form of a relative covariance matrix for 49-neutron energy groups (LB = 5).

Eigenvalues of the relative covariance matrix given in File-33 were calculated by means of PADE-2 and tested in additional by the COVEIG code [3.73]. Six-digit eigenvalues of the relative covariance matrix in File-33 are as follows:

```
3.93887E-06    3.97279E-06    4.03010E-06    4.11008E-06
4.21058E-06    4.34198E-06    4.50826E-06    4.73271E-06
4.99108E-06    5.26634E-06    5.64828E-06    6.19283E-06
6.76888E-06    7.48653E-06    8.90571E-06    9.95649E-06
1.26833E-05    1.44721E-05    1.83550E-05    2.35642E-05
2.69507E-05    3.23243E-05    3.98828E-05    4.80888E-05
5.64407E-05    6.42256E-05    6.99560E-05    7.58638E-05
8.33375E-05    9.09277E-05    9.82055E-05    1.05070E-04
1.11502E-04    1.17486E-04    1.23058E-04    1.28258E-04
1.34099E-04    1.47548E-04    2.74750E-04    1.75742E-03
6.54578E-03    1.05815E-02    1.72985E-02    2.46579E-02
3.68054E-02    4.72128E-02    6.91497E-02    3.71170E-01
5.03713E-01
```

All of these eigenvalues are positive.

Evaluated group cross sections and their uncertainties for the excitation function of the ^{169}Tm(n,3n)^{167}Tm reaction are listed in Table 10.1. Group boundaries are the same as in File-33.

Uncertainties in the evaluated $^{169}Tm(n,3n)^{167}Tm$ excitation function range from 4.91% to 68.99%. One can see from Table 10.1 that the smallest uncertainties in the evaluated cross sections of 4.91% to 4.97% are observed in the neutron energy range from 21.5 to 23.0 MeV. The highest uncertainty in cross sections, 68.99%, is obtained near threshold. Significant uncertainties in cross sections exceeding 10% are between 15.5 - 16.0 MeV and above 30 MeV.

TABLE 10.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{169}Tm(n,3n)^{167}Tm$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 50 MeV.

Neutron energy (MeV) from to	**Cross section (mb)**	**Uncer-tainty (%)**	**Neutron energy (MeV) from to**	**Cross section (mb)**	**Uncer-tainty (%)**
14.963 - 15.500	5.503	68.99	28.000 - 29.000	1264.610	9.05
15.500 - 16.000	18.444	18.90	29.000 - 30.000	1107.686	9.73
16.000 - 16.500	58.596	8.99	30.000 - 31.000	957.764	10.19
16.500 - 17.000	141.026	7.98	31.000 - 32.000	826.534	10.56
17.000 - 17.500	271.346	7.69	32.000 - 33.000	717.484	10.97
17.500 - 18.000	442.393	7.51	33.000 - 34.000	629.383	11.43
18.000 - 18.500	631.494	7.21	34.000 - 35.000	559.112	11.90
18.500 - 19.000	811.153	6.72	35.000 - 36.000	503.242	12.32
19.000 - 19.500	962.985	6.18	36.000 - 37.000	458.708	12.67
19.500 - 20.000	1082.398	5.74	37.000 - 38.000	423.003	12.96
20.000 - 20.500	1174.070	5.44	38.000 - 39.000	394.158	13.20
20.500 - 21.000	1245.724	5.23	39.000 - 40.000	370.657	13.42
21.000 - 21.500	1304.370	5.06	40.000 - 41.000	351.344	13.63
21.500 - 22.000	1355.034	4.95	41.000 - 42.000	335.335	13.86
22.000 - 22.500	1400.690	4.91	42.000 - 43.000	321.952	14.11
22.500 - 23.000	1442.626	4.97	43.000 - 44.000	310.676	14.39
23.000 - 23.500	1480.738	5.12	44.000 - 45.000	301.103	14.70
23.500 - 24.000	1513.892	5.33	45.000 - 46.000	292.917	15.04
24.000 - 24.500	1540.154	5.59	46.000 - 47.000	285.870	15.40
24.500 - 25.000	1557.158	5.87	47.000 - 48.000	279.766	15.78
25.000 - 25.500	1562.438	6.17	48.000 - 49.000	274.447	16.17
25.500 - 26.000	1553.954	6.51	49.000 - 50.000	269.786	16.57
26.000 - 26.500	1530.208	6.91	50.000 - 55.000	259.165	17.71
26.500 - 27.000	1491.686	7.37	55.000 - 60.000	247.194	19.59
27.000 - 28.000	1407.419	8.10			

Fig. 10.1 shows the evaluated excitation function for the $^{169}Tm(n,3n)^{167}Tm$ reaction over the neutron energy range from threshold to 60.0 MeV in comparison with TENDL-2012, EAF-2010, MENDL-2 and experimental data. The MENDL-2 evaluation in the neutron energies around the maximum of the excitation function gives the highest cross section values in comparison with other evaluations and experimental data. The present evaluation as is seen in Fig. 10.1 and especially in Fig. 10.2 agrees better with corrected experimental data than the TENDL-2012, EAF-2010, MENDL-2 libraries.

Unfortunately integral experimental data for the $^{169}Tm(n,3n)^{167}Tm$ reaction are not presented in EXFOR library and in the open publications. The $^{169}Tm(n,3n)^{167}Tm$ reaction average cross section for ^{235}U thermal fission neutron spectrum calculated from the present evaluation and TENDL-2012, EAF-2010, MENDL-2 excitation functions are compared in Table 10.2. Calculation of the averaged cross sections for JENDL-4.0 and JEFF-3.1/A data were not performed because the microscopic cross sections for the $^{169}Tm(n,3n)^{167}Tm$ reaction in those libraries are evaluated only up to 20 MeV. Calculation of the averaged cross section over ^{252}Cf spontaneous fission neutron spectrum was not carried out because this spectrum is evaluated only up to 20 MeV.

TABLE 10.2. CALCULATED INTEGRAL CROSS SECTIONS FOR THE $^{169}Tm(n,3n)^{167}Tm$ REACTION IN ^{235}U THERMAL FISSION NEUTRON SPECTRUM.

Library	**Calculated integral cross section, mb**	**90%-Response range, MeV**
Present evaluation	4.0108E-03	16.1 – 21.5
TENDL-2012	5.9457E-03	15.7 – 21.0
EAF-2010	4.1795E-03	16.1 – 21.5
MENDL-2	3.7488E-03	16.3 – 21.8

It is necessary to note that averaged over ^{235}U thermal fission neutron spectrum maximal cross section (TENDL-2012) and minimal cross section (MENDL-2) are differing in the limit of 58.6%.

Information about 90%-response range given in the Table 10.2 permits to understand in which energy range the $^{169}Tm(n,3n)^{167}Tm$ reaction excitation function may be tested using experimental data for the ^{235}U thermal fission neutron spectrum.

Be(d,n) reaction is one of the best benchmark neutron fields for testing neutron induced reactions with a high threshold. Precise measurements of the integral cross section of the $^{169}Tm(n,3n)^{167}Tm$ reaction in a well characterised neutron spectrum from Be(d,n) reaction are required for a final testing of the evaluated excitation function. Measurements of the integral cross section of the $^{169}Tm(n,3n)^{167}Tm$ reaction for the ^{235}U thermal fission neutron spectrum will be also useful for testing the evaluated excitation function.

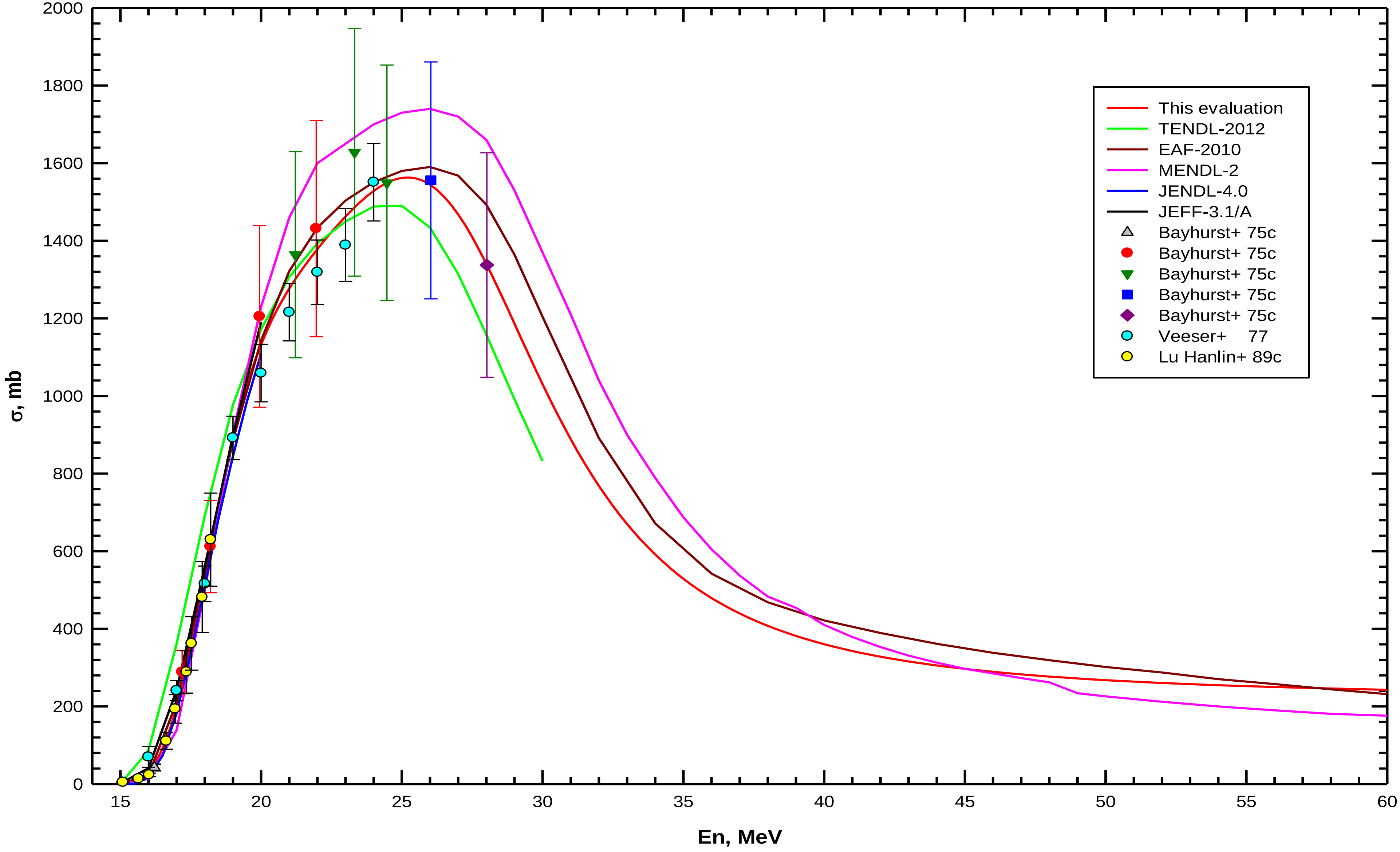

FIG. 10.1. Evaluated excitation function of the ^{169}Tm(n,3n)^{167}Tm reaction in the energy range from threshold to 60 MeV in comparison with TENDL-2012, EAF-2010, MENDL-2, JENDL-4.0, JEFF-3.1/A and experimental data.

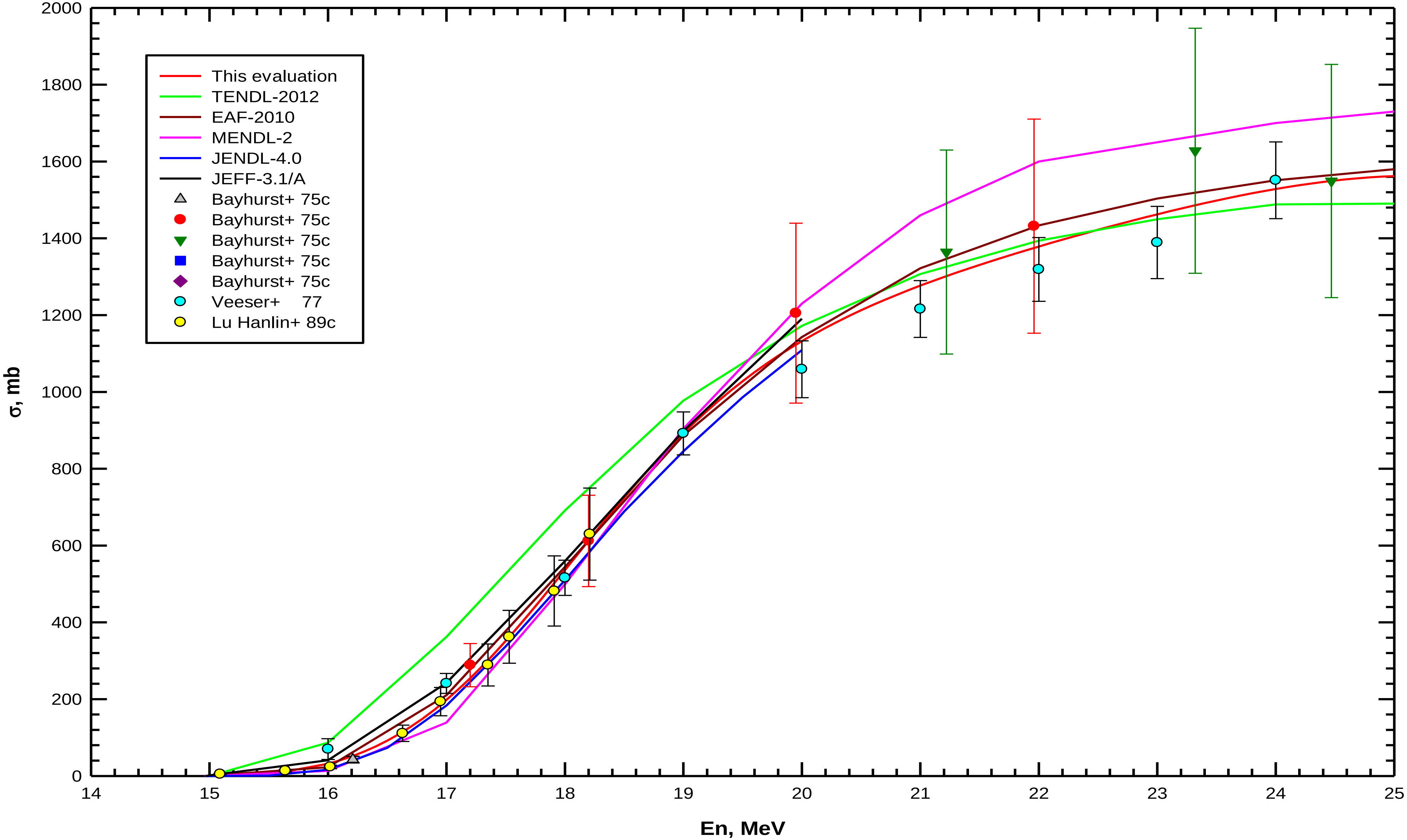

FIG. 10.2. Evaluated excitation function of the $^{169}Tm(n,3n)^{167}Tm$ reaction in the energy range from threshold to 25 MeV in comparison with TENDL-2012, EAF-2010, MENDL-2, JENDL-4.0, JEFF-3.1/A and experimental data

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11. CONCLUSIONS

New evaluations of cross sections and their uncertainties have been carried out for eight dosimetry reactions: ^{54}Fe(n,p)^{54}Mn, ^{58}Ni(n,2n)^{57}Ni, ^{67}Zn(n,p)^{67}Cu, ^{92}Mo(n,p)^{92m}Nb, ^{93}Nb(n,γ)^{94}Nb, ^{113}In(n,n')^{113m}In, ^{115}In(n,γ)^{116m}In and ^{169}Tm(n,3n)^{167}Tm. The Q-values of the ^{54}Fe(n,p)^{54}Mn, ^{67}Zn(n,p)^{67}Cu and ^{92}Mo(n,p)^{92m}Nb reactions are positive and equal to 8.52600E+04 eV, 2.20690E+05 eV and 4.25680E+05 eV. Excitation functions of ^{54}Fe(n,p)^{54}Mn, ^{67}Zn(n,p)^{67}Cu and ^{92}Mo(n,p)^{92m}Nb reactions were evaluated over neutron energy regions (0.7 - 30) MeV, (1 – 50) MeV and (1 – 40) MeV, respectively. Excitation function of the ^{113}In(n,n')^{113m}In dosimetry reaction was evaluated in the neutron energy interval from threshold to 20 MeV. The ^{58}Ni(n,2n)^{57}Ni and ^{169}Tm(n,3n)^{167}Tm reaction excitation functions were evaluated in the energy range from threshold to 60 MeV. Cross sections of the ^{93}Nb(n,γ)^{94}Nb and ^{115}In(n,γ)^{116m}In reactions were evaluated in a wide energy region 1.00E-05 eV - 20 MeV.

In describing the new evaluation the ^{93}Nb(n,γ)^{94}Nb reaction cross section is given in the energy range 1.0E-05 eV – 7.5 keV via the Reich-Moore resonance parameters. The obtained resonance parameters permit to calculate the ^{93}Nb(n,tot), ^{93}Nb(n,el) and ^{93}Nb(n,γ)^{94}Nb excitation functions in the neutron energy range 1.0E-05 eV – 7.5 keV without any additional data (background). In the RRDFF-2012 (ENDF/B-VII.1) library the ^{93}Nb(n,γ)^{94}Nb reaction cross section in the resolved resonance region a significant background is introduced.

In the new evaluation of the ^{115}In(n,γ)^{116m}In reaction excitation function in comparison with equivalent data from the RRDFF-2012 (ENDF/B-VII.1) library, the new evaluation more correctly gives the uncertainty of cross sections and isomeric ratios $R_g(E) = \sigma_g(E)/\sigma_{m+g}(E)$, $R_m(E) = \sigma_m(E)/\sigma_{m+g}(E)$.

Excitation functions of the ^{67}Zn(n,p)^{67}Cu, ^{92}Mo(n,p)^{92m}Nb, ^{113}In(n,n')^{113m}In and ^{169}Tm(n,3n)^{167}Tm reactions were for the first time evaluated for dosimetry applications. Uncertainties in the cross sections for all new evaluations are given in the form of relative covariance matrices.

Benchmark calculations performed for ^{54}Fe(n,p)^{54}Mn, ^{58}Ni(n,2n)^{57}Ni, ^{67}Zn(n,p)^{67}Cu, ^{92}Mo(n,p)^{92m}Nb, ^{113}In(n,n')^{113m}In, ^{115}In(n,γ)^{116m}In reactions using the ^{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 TENDL-2010, EAF-2010, JEFF-3.1/A, JENDL-4.0, JENDL/HE and MENDL-2 libraries. Presented in this work is the newly evaluated excitation function for the ^{169}Tm(n,3n)^{167}Tm reaction which is also better agreeing with related experimental data than the equivalent data from TENDL-2010, EAF-2010 and MENDL-2 libraries. Thus, the ^{54}Fe(n,p)^{54}Mn, ^{58}Ni(n,2n)^{57}Ni, ^{67}Zn(n,p)^{67}Cu, ^{92}Mo(n,p)^{92m}Nb, ^{93}Nb(n,γ)^{94}Nb, ^{113}In(n,n')^{113m}In, ^{115}In(n,γ)^{116m}In and ^{169}Tm(n,3n)^{167}Tm cross section files in ENDF-6 format should be considered as suitable candidates in the preparation of an improved version of the International Dosimetry File IRDFF-2012 (IRDFF v1). These entire evaluated neutron data may also be recommended for inclusion in activation data files for inventory calculations in different types of reactors.

Precise measurements of the integral cross section of the ^{169}Tm(n,3n)^{167}Tm reaction in a well determined neutron spectra from Be(d,n) reaction are required for a final testing of the excitation function for this reaction. New measurements and evaluation of ^{167}Tm decay parameters are also required. First of all this refers to a more exact determination of the probability of 207.8-keV gamma radiation accompanying decay of ^{167}Tm.

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Nuclear Data Section
International Atomic Energy Agency
Vienna International Centre, P.O. Box 100
A-1400 Vienna, Austria

E-mail: nds.contact-point@iaea.org
Fax: (43-1) 26007
Telephone: (43-1) 2600 21725
Web: http://www-nds.iaea.org