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Evaluation of ^{23}Na(n,γ)^{24}Na, ^{23}Na(n,2n)^{22}Na and ^{27}Al(n,2n)^{26}Al reaction cross sections for the IRDFF library

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August 2017

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1. INTRODUCTION

Cross-section data for the $^{23}Na(n,2n)^{22}Na$, $^{23}Na(n,\gamma)^{24}Na$, and $^{27}Al(n,2n)^{26}Al$ reactions are needed to solve a wide spectrum of scientific and technical tasks. Activation detectors based on the $^{23}Na(n,2n)^{22}Na$ and $^{23}Na(n,\gamma)^{24}Na$ reactions are used in the field of reactor dosimetry. The production of the long-lived radionuclide ^{26}Al is of considerable interest for fusion technology. The threshold of the $^{27}Al(n,2n)^{26g}Al$ reaction (E_{th} = 13.55 MeV) is very close to the energy of neutrons generated in deuterium-tritium fusion plasma – 14 MeV. The sensitivity of ^{26g}Al activity on the temperature-dependent neutron energy distribution leads to the possibility of utilizing the $^{27}Al(n,2n)^{26g}Al$ reaction for plasma temperature measurement.

In the current version of the International Reactor Dosimetry and Fusion File – IRDFF-v1.05 [1.1], data for the $^{23}Na(n,2n)^{22}Na$ reaction excitation function were obtained from two sources. Cross sections and related uncertainties in the neutron energy range from threshold to 20 MeV were included from JENDL/D-99 library [1.2] and were evaluated by S. Iwasaki. Extended cross sections and covariances from 20 to 60 MeV were obtained by A. Trkov on the basis of TENDL-2011 data. The $^{23}Na(n,2n)^{22}Na$ reaction excitation function from TENDL-2011 was renormalized at 20 MeV to JENDL/D-99 data.
The excitation function for the $^{23}Na(n,\gamma)^{24}Na$ reaction presented in the IRDFF-v1.5 was taken from the ENDF/B-VI library, MOD 2 Revision, September 2000 [1.3]. This evaluation was also adopted in the next version of the American library - ENDF/B-VII.1.

Data for the $^{27}Al(n,2n)^{26}Al$ reaction excitation function are absent in the IRDFF-v1.05 library. Data for this reaction are also absent in the JENDL/D-99 dosimetry file [1.2]. Excitation functions of partial $^{27}Al(n,2n)^{26m}Al$ and $^{27}Al(n,2n)^{26g}Al$ reactions are presented in the TENDL-2014 [1.3] and EAF-2010 [1.4] libraries. It is necessary to mark that the $^{27}Al(n,2n)^{26}Al$ reaction cross sections in the ENDF/B-VII.1, JEFF-3.1/A, JENDL-4.0 and all the rest libraries were evaluated as total data up to 20 MeV and are not appropriate for reactor and fusion dosimetry application.

The main aim of this work was the evaluation of cross-section data and related covariance matrixes of uncertainty for $^{23}Na(n,2n)^{22}Na$, $^{23}Na(n,\gamma)^{24}Na$, and $^{27}Al(n,2n)^{26}Al$ reactions. The excitation function for the $^{23}Na(n,2n)^{22}Na$ and $^{27}Al(n,2n)^{26}A$ reactions needs to be evaluated with extension to higher neutron energies up to 60 MeV. New evaluations were performed on the basis of all available experimental data corrected to the new standards ???? and data obtained from consistent theoretical model calculations.

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2. METHOD OF EVALUATION OF THE REACTION EXCITATION FUNCTIONS

2.1. Sources of information used in the evaluation

In the process of evaluating cross sections and their uncertainties, two common information sources were used for reactions $^{23}Na(n,\gamma)^{24}Na$, $^{23}Na(n,2n)^{24}Na$ $^{27}Al(n,2n)^{26m}Al$ and $^{27}Al(n,2n)^{26g}Al$: available differential and integral experimental data. Differential and integral experimental data were mainly taken from the EXFOR Library (Version November 2015). In those cases where the necessary information was absent in EXFOR, the information was taken from the original publications.

2.2. Analysis of experimental data

In the first step of evaluation, all experimental data were analysed. During this procedure all experimental data, where possible, were corrected to the new recommended cross section data for monitor reactions used in the measurements and to the new recommended decay data.
The correction of experimental data to the new standards, in general, leads to a decrease of discrepancies in the experimental data and thus to a decrease of the uncertainty in the evaluated cross section values. The information needed about standards used for the correction of microscopic experimental data under investigation is given in the Table 2.1.

TABLE 2.1. DATA USED AS STANDARDS FOR CORRECTION OF MICROSCOPIC EXPERIMENTAL CROSS SECTIONS.

Monitor reaction	Cross section used as standard		Half-life of residual nuclei	Radiation mode and energy, keV		Emission Probability per decay	
$^{1}H(n,n)^{1}H$	Carlson+	[2.1]					
$^{6}Li(n,t)^{4}He$	Carlson+	[2.1]					
$^{27}Al(n,\alpha)^{24}Na$	Zolotarev	[2.2]	14.997 (12) h	Gamma	1368.63	0.999936 (15)	[2.12]
				Gamma	2754.01	0.99855 (5)	[2.12]
$^{27}Al(n,p)^{27}Mg$	Zolotarev+	[2.3]	9.458 (12) m	Gamma	843.76	0.718 (4)	[2.13]
				Gamma	1014.44	0.280 (4)	[2.13]
$^{54}Fe(n,p)^{54}Mn$	Zolotarev+	[2.4]	312.05 (4) d	Gamma	834.848	0.99976 (1)	[2.14]
$^{56}Fe(n,p)^{56}Mn$	Zolotarev	[2.5]	2.5789 (1) h	Gamma	846.76	0.9887 (3)	[2.13]
				Gamma	1810.73	0.2719 (79)	[2.13]
$^{58}Ni(n,x)^{57}Co$	Zolotarev	[2.6]	271.74 (6) d	Gamma	122.06	0. 8560 (17)	[2.15]
				Gamma	136.47	0.1068 (8)	[2.15]
$^{63}Cu(n,2n)^{62}Cu$	Zolotarev	[2.7]	9.67 (3) m	Beta+	2936.9	0.9760 (3)	[2.16]
				Gamma	511	1.9566 (5)	[2.16]
				Gamma	1172.97	0.00342 (5)	[2.16]
$^{65}Cu(n,2n)^{64}Cu$	Zolotarev	[2.7]	12.700 (2) h	Beta+	653.1	0.1740 (22)	[2.17]
				Beta-	578.7	0.390 (4)	[2.17]
				Gamma	511	0.348 (4)	[2.17]
				Gamma	1345.77	0.00473 (10)	[2.17]
$^{75}As(n,2n)^{74}As$	Zolotarev	[2.8]	17.77 (2) d	Gamma	511	0.580 (60)	[2.18]
$^{93}Nb(n,2n)^{92m}Nb$	Zolotarev	[2.9]	10.15 (2) d	Gamma	934.44	0.9907 (4)	[2.13]

Monitor reaction	Cross section used as standard	Half-life of residual nuclei	Radiation mode and energy, keV	Emission Probability per decay
$^{115}In(n,\gamma)^{116m}In$	Zolotarev+ [2.4]	54.29 (17) m	Gamma 416.90 Gamma 1097.28 Gamma 1293.56	0.272 (4) [2.19] 0.585 (8) [2.19] 0.848 (12) [2.19]
$^{127}I(n,\gamma)^{128}I$	Zolotarev [2.10]	24.99 (2) m	Gamma 442.90	0.169 (17) [2.13]
$^{141}Pr(n,2n)^{140}Pr$	Zolotarev [2.11]	3.39 (1) m	Gamma 511	1.020 (6) [2.20]
$^{197}Au(n,2n)^{196}Au$	Zolotarev [2.7]	6.183 (10) d	Gamma 333.03 Gamma 355.73 Gamma 426.10	0.229 (6) [2.13] 0.870 (4) [2.13] 0.066 (4) [2.13]
$^{197}Au(n,\gamma)^{198}Au$	Carlson+ [2.1]	2.6947 (3) d	Gamma 411.802	0.9562 (4) [2.21]
$^{235}U(n,f)$	Carlson+ [2.1]			

Comment for Table 2.1: for beta transition the end-point value of energy is given.

Recommended thermal neutron cross sections at 0.0253 eV were taken mainly from works [2.1] and [2.22]. For the problematic reactions $^{31}P(n,\gamma)^{32}P$, $^{34}S(n,\gamma)^{35}S$, $^{75}As(n,\gamma)^{76}As$ cross sections at 0.0253 eV were also taken from Refs. [2.40-2.42]. The value of $^{23}Na(n,\gamma)^{24}Na$ cross section at 0.0253 eV was taken from the new evaluation described in this work.

Recommended cross section and decay data for monitor reactions used in the measurements of integral cross sections in ^{235}U thermal fission neutron spectrum and ^{252}Cf spontaneous fission neutron spectrum are given below in Table 2.2.

TABLE 2.2. DATA USED AS STANDARDS FOR CORRECTION OF INTEGRAL EXPERIMENTAL CROSS SECTIONS MEASURED IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Monitor Reaction	Cross section used as standard, mb	Half-life for residual nuclei	Radiation mode and energy, keV	Emission Probability per decay
$^{27}Al(n,\alpha)^{24}Na$	0.7007±1.28% [2.23] U 1.016±1.28% [2.24] Cf	14.997 (12) h	Gamma 1368.63 Gamma 2754.01	0.999936(15) [2.12] 0.99855 (5) [2.12]
$^{54}Fe(n,p)^{54}Mn$	78.09±1.50% [2.4] U	312.05 (4) d	Gamma 834.848	0.999760(10) [2.14]
$^{56}Fe(n,p)^{56}Mn$	1.079±1.54% [2.23] U	2.5789 (1) h	Gamma 846.754 Gamma 1810.72	0.9887 (3) [2.13] 0.2719 (79) [2.13]
$^{58}Ni(n,p)^{58}Co$	108.2±1.30% [2.23] U	70.86 (6) d	Gamma 511 Gamma 810.78	0.298 (4) [2.25] 0.99450 (10) [2.25]
$^{115}In(n,n')^{115m}In$	187.8±1.23% [2.23] U 197.4±1.37% [2.24] Cf	4.486 (4) h	Gamma 336.24	0.458 (22) [2.26]
$^{238}U(n,f)$	325.7±1.64% [2.24] Cf			

Comments for Table 1.2: Symbol "U" – means ^{235}U thermal fission neutron spectrum,
Symbol "Cf" – means ^{252}Cf spontaneous fission neutron spectrum.

Digital data for ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were taken from IRDFF-v1.05 neutron spectra data file material numbers MAT=9228 and MAT=9861, respectively. Neutron spectra data files were prepared by A. Trkov. Both spectra were taken from the ENDF/B-VII decay data library and extrapolated to 30 MeV. The interpolation laws were refined to avoid bumps in the spectra (A. Trkov, November 2013).
Information about isotopic compositions of the elements was taken from Ref. [2.27].

2.3. Theoretical model calculation cross-section values for dosimetry reactions

The theoretical model calculation was used as additional source of cross-section information for reactions with poor experimental data. In our case, the theoretical model calculation was carried out for the determination of the relative shape $^{23}Na(n,\gamma)^{24}Na$ and $^{23}Na(n,2n)^{22}Na$ reactions excitation functions above 2 MeV and 20 MeV, respectively.

For the theoretical description of the excitation functions the optical-statistical method was used while taking into account consistently the contribution of the direct, pre-equilibrium and statistical equilibrium processes in the different outgoing channels.

The practical calculations of the cross sections were made by means of a modified version of the GNASH code [2.28]. The modified GNASH code is differing in general from the original GNASH code [2.29]. The modified version includes subroutine which permits to take into account the neutron width fluctuation. Furthermore, the modified GNASH code has a mode which permits to calculate the cross-section of population of individual levels excited in the investigated reaction. This capability is very important to calculate the $^{27}Al(n,2n)^{26g}Al$ and $^{27}Al(n,2n)^{26m}Al$ reaction cross sections.

The calculation of penetrability coefficients for neutrons was made on the basis of a generalized optical model, which permits to estimate the cross sections for the direct excitations of collective low-lying levels. The ECIS coupled channel deformed optical model code [2.30] was used for these calculations. The optical coefficients of proton and alpha particle penetrabilities were determined by means of the SCAT2 code [2.31].

The data file used by the GNASH code on the discrete levels parameters of residual nuclei was formed on the basis data given in Ref. [2.13] and permanently corrected. 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 in accordance with the radiation strength functions recommended in Ref. [2.32].

Continuum level densities were represented with the Gilbert-Cameron [2.33] model using the Cook parameters [2.34] (mode IBSF=1 in the GNASH code). The calculation of gamma-ray transition probabilities in the continuum region of excited states of all nuclei under consideration was made in the frame of hypothesis of domination of giant dipole resonance with parameters of the radiative strength function from the Kopecky-Uhl systematic [2.35]. Recommended parameters of giant dipole resonances were taken from Ref. [2.36].

2.4. Statistical analysis of cross-sections from the database

The method of statistical analysis of correlated data used to evaluate the excitation functions of $^{23}Na(n,2n)^{22}Na$, $^{27}Al(n,2n)^{26m}Al$ and $^{27}Al(n,2n)^{26g}Al$ reactions from threshold to 60 MeV and $^{23}Na(n,\gamma)^{24}Na$ reaction in the energy range 459.3 keV – 20 MeV was described in the IAEA NDS Reports [2.2-2.4], [2.7], [2.9]. Detailed description of the method and PADE-2 code may be found in Refs. [2.37-2.39].

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3. RE-EVALUATION OF THE $^{23}Na(n,\gamma)^{24}Na$ REACTION EXCITATION FUNCTION

The abundance of ^{23}Na isotope in natural sodium is 100 atom percent. The (J_π=4+) ground level of ^{24}Na obtained in the (n,γ) reaction undergoes 100% via β^- decay mode with a half-life of (14.997 ± 0.012) hours. The β^- decay is accompanied by emission of beta- particles and gamma-ray radiation. The 554.1 keV β^- rays with end-point energy of 1392.56 keV is at its most intense in beta-spectrum, I_{β^-}= 0.99855 ± 0.00005. Mean energy of emitted beta-particles: 554.6 keV, total beta- intensity: 0.99995 ± 0.00008. The gamma-lines with energies 1368.626-keV (I_γ = 0.999936 ± 0.000015) and 2754.007-keV (I_γ = 0.99855 ± 0.00005) are normally used to determine the $^{23}Na(n,\gamma)^{24}Na$ reaction rate. Recommended decay data for the half-life and emission probabilities I_{β^-}, I_γ per decay of ^{24}Na were taken from Ref. [3.45].

Microscopic experimental data for the $^{23}Na(n,\gamma)^{24}Na$ reaction excitation function is given in [3.1-3.44]. The larger part of cross-section measurements was carried out in a region of thermal neutrons [3.1-3.11], [3.15], [3.20-3.24], [3.27], [3.33-3.34], [3.36], [3.38], [3.40-3.44]. The rest of the work gives experimental information about cross sections from 2 keV to 19.39 MeV. Experimental data of [3.1], [3.3-3.7], [3.9-3.13], [3.15-3.20], [3.23-3.33] and [3.36-3.39] were, in the process of analysis, corrected to the new standards for the relevant monitor reactions (see Table 2.1) and to the recommended decay data for ^{24}Na.

In the described re-evaluation of the $^{23}Na(n,\gamma)^{24}Na$ reaction excitation function, theyaho cross section in the energy range 1.0000E-05 eV – 459.3 keV is given via the Reich-Moore (RM) resonance parameters. The upper boundary of the resolved resonance region is limited due to the threshold of inelastic scattering. Resonance parameters evaluated in the JEFF-3.2 library [3.46] were used as preliminary source of information.

For the re-evaluation of ^{23}Na resonance parameters, all available experimental information for resonance parameters and cross sections for the $^{23}Na(n,tot)$, $^{23}Na(n,el)$ and $^{23}Na(n,\gamma)$ reactions in the resolved resonance region was analyzed. The analyzed results of measurements of the ^{23}Na resonance parameters include works [3.34], [3.50-3.52], [3.60-3.74].
Experimental data for the total neutron cross section on ^{23}Na in the resolved resonance region are presented in works [3.47-3.55]. Experimental cross section data for the $^{23}Na(n,el)$ reaction in the resonance energy range are given in [3.47], [3.54], [3.56-3.59]. The $^{23}Na(n,\gamma)$ cross sections in the RR region were obtained in [3.1-3.11], [3.12-3.13], [3.15], [3.16-3.19], [3.20-3.24], [3.25-3.27], [3.32-3.34], [3.36], [3.38], [3.40-3.44].

As mentioned above, the larger part of $^{23}Na(n,\gamma)^{24}Na$ cross-section measurements was carried out in the region of thermal neutrons. Experimental cross-section data at thermal region had been measured by five methods: transmission, pile oscillator method, classical activation method, k_0- activation method, and method of prompt gamma-ray analysis (PGAA). The $^{23}Na(n,\gamma)^{24}Na$ experimental data at 0.0253 eV are compared in Table 3.1.

The k_0- method was developed especially for reactor neutron activation analysis (NAA). This method permitted to standardize NAA measurements at reactor spectra with different ratio of the thermal neutron flux to epithermal flux. Thermal cross sections at 0.0253 eV are determined from measured $k_{0,m}$, $Q_{0,m}$ and $Q_{0,x}$ - factors, where $Q_{0,m}$ and $Q_{0,x}$ are ratios of resonance integrals to 2200 ms^{-1} cross-section ratio for the monitor reaction and investigated reaction, respectively. Usually, the $^{197}Au(n,\gamma)^{198}Au$ reaction is used as the monitor in the k_0-activation method..

TABLE 3.1. COMPARISON OF $^{23}Na(n,\gamma)^{24}Na$ EXPERIMENTAL CROSS SECTION DATA AT 0.0253 EV.

Author	Neutron source	Method	Monitor reaction and ref to standardization value	Cross section, mb
Coltman+ [3.1]	Radium-alpha-beryllium source in a paraffin block	Transmission	B00(n,abs) [2.22]	601.0 ± 51.2
Seren+ [3.2]	Reactor at ANL, thermal spectrum	Activation		630 ± 126
Pomerance [3.3]	thermal spectrum	Pile oscillator	$^{197}Au(n,\gamma)^{198}Au$ [2.1]	488.0 ± 24.4
Grimeland [3.4]	KJELLER reactor, thermal column	Activation	$^{127}I(n,\gamma)^{128}I$ [2.22]	574.0 ± 45.6
Bartholomew+[3.5]	Reactor at Chalk River	Activation	$^{197}Au(n,\gamma)^{198}Au$ [2.1]	562.0 ± 33.9
Harris+ [3.6]	Reactor at ANL	Pile oscillator	B00(n,abs) [2.22]	511.0 ± 5.1
Popovic+ [3.7]	JEEP reactor, Thermal column	Activation	$^{233}U(n,f)$ [2.22]	537.2 ± 16.2
Groshev+ [3.8]	Thermal column of the RFT reactor	Activation	No information	500.0 ± 50.0
Brooksbank+ [3.9]	Reactor at ORNL	Activation	$^{55}Mn(n,\gamma)^{56}Mn$ [2.22]	502.0 ± 50.2
Grimeland [3.10]	KJELLER reactor, thermal column	Activation	B00(n,abs) [2.22]	521.6 ± 31.1
Cocking+ [3.11]	Reactor BEPO at AERE	Activation	$^{197}Au(n,\gamma)^{198}Au$ [2.1]	536.9 ± 6.1
Rose+ [3.15]	Reactors DIMPLE and GLEEP at AERE	Pile oscillator	B00(n,abs) [2.22]	536.3 ± 9.8
Wolf [3.20]	FRM-reactor of Munich	Activation	$^{197}Au(n,\gamma)^{198}Au$ [2.3]	530.8 ± 8.0
Meadows+ [3.21]	Moderated neutrons from Li(p,n) reaction, thermal spectrum	Pile oscillator		470.0 ± 60.0
Jozefowicz [3.23]	Thermal column of EWA reactor	Activation	$^{55}Mn(n,\gamma)^{56}Mn$ [2.22] $^{31}P(n,\gamma)^{32}P$ [2.22] $^{31}P(n,\gamma)^{32}P$ [2.42] $^{127}I(n,\gamma)^{128}I$ [2.22]	534.4 ± 12.3 507.7 ± 19.4 529.2 ± 25.7 569.4 ± 19.6
Koehler [3.24]	FRM reactor, water poll	Activation	$^{197}Au(n,\gamma)^{198}Au$ [2.1]	499.0 ± 20.0
Kappe [3.27]	Reactor, thermal spectrum	Activation	$^{31}P(n,\gamma)^{32}P$ [2.22] $^{31}P(n,\gamma)^{32}P$ [2.42] $^{34}S(n,\gamma)^{35}S$ [2.22] $^{34}S(n,\gamma)^{35}S$ [2.43] $^{37}Cl(n,\gamma)^{38}Cl$ [2.22] $^{75}As(n,\gamma)^{76}As$ [2.22] $^{75}As(n,\gamma)^{76}As$ [2.44] $^{127}I(n,\gamma)^{128}I$ [2.22]	478.3 ± 16.8 498.6 ± 22.9 577.9 ± 30.8 558.2 ± 29.9 528.0 ± 12.9 511.0 ± 14.3 548.9 ± 17.3 577.0 ± 21.9
	Be(d,n) NPL	Activation	$^{197}Au(n,\gamma)^{198}Au$ [2.1]	526.2 ± 4.3

Author	Neutron source	Method	Monitor reaction and ref to standardization value	Cross section, mb
Ryves+ [3.33]	standart thermal neutron source			
Yamamuro+ [3.34]	Linear accelerator, Ta photoneutrons	Transmission	Cd00(n,γ) [2.22]	500.0 ± 30.0
Gleason [3.36]	Graphite-moderated Cf-252 facility	Activation	^{197}Au(n,γ)^{198}Au [2.1] ^{55}Mn(n,γ)^{56}Mn [2.22]	539.0 ± 20.0
Heft [3.38]	LPTR Research reactor facility at LRL	Activation	^{45}Sc(n,γ)^{46}Sc [2.22]	523.0 ± 5.0
Kaminishi+ [3.40]	Am-241 Alpha-Be source, moderated with paraffin	Activation		577.0 ± 8.0
Kennedy+ [3.41]	Reactor	K_0 - method	^{197}Au(n,γ)^{198}Au [2.1]	515.0 ± 12.9
Szentmiklosi [3.42]	Cold neutron beams at the Budapest reactor	PGAA	^{197}Au(n,γ)^{198}Au [2.1]	527.0 ± 8
Farina Arb+ [3.43]	BR1 reactor at SCK-CEN, Iradiation at 4 positions	K_0 - method	^{197}Au(n,γ)^{198}Au [2.1]	519.1 ± 8.2
Firestone+ [3.44]	Cold neutron beams at the Budapest reactor	PGAA	^{35}Cl(n,γ)^{36}Cl γ-ray [3.44]	541.0 ± 3

Measured values of ^{23}Na(n,γ)^{24}Na thermal cross section range from (470 ± 60) mb [3.21] to (630 ± 126) mb [3.2]. Measurements [3.7], [3.10], [3.11], [3.15], [3.20], [3.23], [3.33], [3.42] carried out in facilities with a well thermalized neutron spectrum and by using well known standards give a value of ^{23}Na(n,γ)^{24}Na cross section at 0.0253 eV between (521.6 – 537.2) mb.

Two resent measurements performed by Farina Arbocco et al. [3.43] and by Firestone et al. [3.44] increased the interval of representative data for 2200 m/s cross section from 519.1 mb to 541.0 mb.

Farina Arbocco et al. determined the ^{23}Na(n,γ)^{24}Na 2200 m/s cross section from measured $k_{0,Au}$ – factors for gamma-lines with energies 1368.626-keV (I_γ = 0.999936 ± 0.000015) and 2754.007-keV (I_γ = 0.99855 ± 0.00005). Measurements were carried out at the BR1 reactor at SCK-CEN, Belgium. Irradiation was performed at 4 positions: channels S84, Y4, X26 and Cavity. The ratio of the thermal neutron flux to epithermal flux – **f** and the slope parameter – **a** of the epithermal spectrum (~1/(E**(1+a)) at the irradiation positions were the following: channel S84 (**f**=16.4±0.4, **a**=0.066±0.003), channel Y4 (**f**=38.2±0.5, **a**=-0.003±0.005), channel X26 (**f**=95±5, **a**=0.11±0.01), Cavity (**f**=70000 , pure thermal). Measured k_0 -values with one-sigma uncertainty are equal to 4.71E-2 (±1.9%) for Eγ=1368.6 keV and 4.72E-2 (±1.6%) for Eγ=2754.0 keV. Obtained by Farina Arbocco et al. k_0 –values are higher than recommended values 4.68E-2 (±0.6%) for Eγ=1368.6 keV and 4.62E-2 (±0.8%) for Eγ=2754.0 keV [3.80].

Assumed parameters for the monitor reaction $^{197}Au(n,\gamma)^{198}Au$ were taken by Authors as follows: <Er>=5.7 eV (±14.2.% 2 sigma), $Q_{0,Au}$ =15.7 (±3.6% 2 sigma), k_0=1.00, F_{Cd}=0.991, $T_{1/2}$=2.695d , Gamma energy 411.8 keV, Intensity 95.54% , $\sigma_{0,Au}$=98.65(9) b.

The thermal neutron cross-section σ_0 (0.0253 eV) for $^{23}Na(n,\gamma)^{24}Na$ reaction was extrapolated with aid of the k0-definition, atomic weight, isotopic composition, the gamma intensities values. In EXFOR 23266004 an average cross section value σ_0 is given obtained from measurements for two gamma lines (519.1 ± 1.0) mb. The assigned uncertainty ±1.0 mb (±0.2%) reflects the discrepancy between the two experimental data 518 mb and 520 mb. The total uncertainty of ±8.2 mb (±0.2%) was calculated from partial uncertainties described in the paper.

Firestone, Revay and Belgia [3.44] obtained the $^{23}Na(n,\gamma)^{24}Na$ reaction thermal neutron cross-section σ_0 (0.0253 eV) in measurements with cold guided neutron beams at the 10 MW Budapest Reactor. In this experiment, the ^{24}Na thermal neutron capture γ-ray cross-sections were measured. Prompt γ-rays from the target were measured with an n-type high-purity, 25% relative efficiency, a germanium (HPGe) detector with closed-end coaxial geometry located 23.5 cm from target. The detector is Compton suppressed by bismuth-germanate (BGO) scintillator guard annulus surrounded by 10-cm-thick lead shielding. The ^{24}Na thermal neutron capture γ-ray cross-sections were internally calibrated with a stoichiometric NaCl target by using as standard $^{35}Cl(n,\gamma)^{36}Cl$ γ-ray cross-sections for gamma energy 1951.1 keV. The assumed parameters for the monitor reaction $^{35}Cl(n,\gamma)^{36}Cl$ were taken by Authors as follows: σ_γ(1951.1 keV)=6.51(2) b [3.81], γ-ray emission probability of Molnar et al. [3.82]. Firestone, Revay and Belgia determined σ_0(0.0253 eV) cross section as from prompt gamma spectrum measurements and decay gamma spectrum measurements. The average value from both measurements is equal to (541±3) mb.

The comparison of two recent measurements performed by Farina Arbocco et al. [3.43] and by Firestone et al. [3.44] shows the inconsistency, (519.1 ± 8.2) mb and (541±3) mb, respectively. The cross section of (541±3) mb determined by Firestone, Revay and Belgia [3.44] is also inconsistent with the σ_0(0.0253 eV) cross section value of (527±8) mb obtained by Szentmiklosi, Revay and Belgia at the same facility by the PGAA method [3.42].

Measurements [3.42] and [3.44] differ mainly in the use of monitor reactions: $^{197}Au(n,\gamma)^{198}Au$ and $^{35}Cl(n,\gamma)^{36}Cl$, respectively. Standard parameters for $^{197}Au(n,\gamma)^{198}Au$ reaction well known. The thermal cross section σ_0(0.0253 eV) obtained in [3.42] is consistent with data by Farina Arbocco et al. [3.43] and shows excellent agreement with precise measurements by Ryves and Perkin [3.33]. This fact permits to propose that recommended parameters for $^{35}Cl(n,\gamma)^{36}Cl$ reaction are not so well determined in the works [3.81-3.82].

It should be noted that Szentmiklosi, Revay and Belgia also measured k_0 -values for Eγ=1368.66 keV, k_0=4.7591E-2 (±2.5%) and for Eγ=2754.13 keV, k_0 = 4.7646E-2 (±2.1%). Measurements of the k_0 -values were carried out in the pure thermal spectrum where the epithermal activation is completely avoided.

The Na24 k_0 -values for Eγ=1368.66 keV, k_0=4.7591E-2 (±2.5%) and for Eγ=2754.13 keV, k_0 = 4.7646E-2 (±2.1%) measured in the pure thermal spectrum are by a factor of 1.01159 and 1.00828 higher than equivalent data by Farina Arbocco et al. [3.43], and by a factor of 1.01808 and 1.03011 higher than recommended values by F. DE CORTE, A. SIMONITS [3.80].

Radiative capture cross section of (530.4±11.2) mb at 0.0253 eV was evaluated from analysis corrected to the new standards and is representative experimental data [3.7], [3.10], [3.11], [3.15], [3.20], [3.23], [3.33], [3.42] and [3.43-3.44].

Evaluated in this work $^{23}Na(n,\gamma)^{24}Na$ cross section of (530.4±11.2) mb at 0.0253 eV in comparison with recommended by S.F. Mughabhab value of (517±4) mb [2.22] is higher by 2.52 % and in excellent agreement with his preliminary value of (530±5) mb [2.42].

In the process of evaluation, the cross sections re-constructed from ^{23}Na resonance parameters for the $^{23}Na(n,tot)$, $^{23}Na(n,el)$ and $^{23}Na(n,\gamma)^{24}Na$ reactions were compared to corresponding experimental data. The tested values of the resonance parameters were: E_r – energy of resonance, L – neutron orbital momentum, J_π – spin and parity of resonance, Γ_n – neutron width, Γ_γ – radiative width.

Evaluation of RM parameters in the energy range 1.000E-5 eV - 32.48 keV was based on the representative experimental data [3.10-3.11], [3.15], [3.33], [3.38] [3.42-3.44], [3.49], [3.51-3.52] and [3.54-3.55].

Parameters of the negative S-resonance: energy of resonance - E_r = -1.760E+5 eV, spin of resonance – J = 2, neutron width - Γ_n = 3.975E+4 eV, radiative width - Γ_γ = 3.697+0 eV were adopted to obtain the $^{23}Na(n,el)$ and $^{23}Na(n,\gamma)$ cross sections at 0.0253 eV equal to (3.038±0.070) barns [2.22] and (530.4±11.2) mb, respectively.

Parameters of the first S-resonance were taken equal to: Er = 2.8465+3 eV, J = 1, GN = 3.7715+2 eV and GG = 3.4000-1 eV.

The neutron orbital values, widths and spins above 32.48 keV were obtained from the careful analysis of Larson's data [3.49]. Gamma widths were evaluated by using the above mentioned $^{23}Na(n,\gamma)^{24}Na$ cross sections measured in the RR region. Gamma widths of 35.355 keV and 53.237 keV resonances were taken from measurements by J. Hori et al. [3.70].
The upper boundary of analyzed resonances was 1.2 MeV. Parameters of resonances with energies above 1.2 MeV were adopted from the JEFF-3.2 library. The scattering radius was taken equal to 4.9 fermi in accordance with the Atlas of Neutron Resonances [2.22].

The evaluated parameters in RM formalism for 22 resonances of 3 l-values are used for reconstruction of $^{23}Na(n,tot)$, $^{23}Na(n,el)$, $^{23}Na(n,\gamma)$ cross sections from 1.000E-5 eV to 4.593E+5 eV. Calculated from RM parameters 2200 m/sec cross sections and resonance integrals are given as follows:

TABLE 3.2. AVERAGE PARAMETERS FROM INTER (T=300K)

Reaction	2200 m/s cross section, barns	Resonance integral, barns
$^{23}Na(n,\gamma)$	5.30400E-01	3.10845E-01
$^{23}Na(n,el)$	3.05382E+00	1.15949E+02
$^{23}Na(n,tot)$	3.58419E+00	1.16259E+02

Reconstructed from the evaluated RM resonance parameters $^{23}Na(n,el)$ and $^{23}Na(n,tot)$ excitation functions in RRR region are shown in Figs. 3.1-3.2 and Figs. 3.3-3.6, respectively, in comparison with equivalent data from IRDFF-v1.5 library and experimental data.

As is seen from Fig. 3.1 the re-evaluated ^{23}Na(n,el) excitation function at 0.0253 eV is passed through the representative experimental data by Koester et al. [3.54], which was adopted by Mughabghab [2.22]. Below 1 keV the IRDFF-v1.5 library gives systematically higher elastic scattering cross sections in comparison with new re-evaluation. In the energy range 0.1 – 0.4593 MeV the re-evaluated ^{23}Na(n,el) excitation function and data from IRDFF-v1.5 library are similar and agree reasonably with experimental data by Korzh and Skljar [3.58] and Chien and Smith [3.59].

The excitation function reconstructed from the evaluated RM resonance parameters for the ^{23}Na(n,tot) reaction in the energy range 1.0E-8 – 1.0E-2 MeV agrees well with the main bulk of experimental data. Presented in Fig. 3.3 old experimental data by Joki et al. [3.53] were renormalized for the representative experimental data by Koester et al. [3.58] in overlapping interval 1.26 – 5.19 eV. As is seen from Fig. 3.3, below 1 keV the IRDFF-v1.5 library gives systematically higher total cross sections in comparison with new re-evaluation. The excitation function reconstructed from the evaluated RM resonance parameters for the ^{23}Na(n,tot) reaction above 32.48 keV agrees well with experimental data of Larson et al.[3.53] up to 0.4593 MeV (see Figs. 3.4-3.6). The excellent agreement between Larson's and IRDFF-v1.5 data in the RRR region is due to the background which was used for description of cross sections in the IRDFF-v1.5 library

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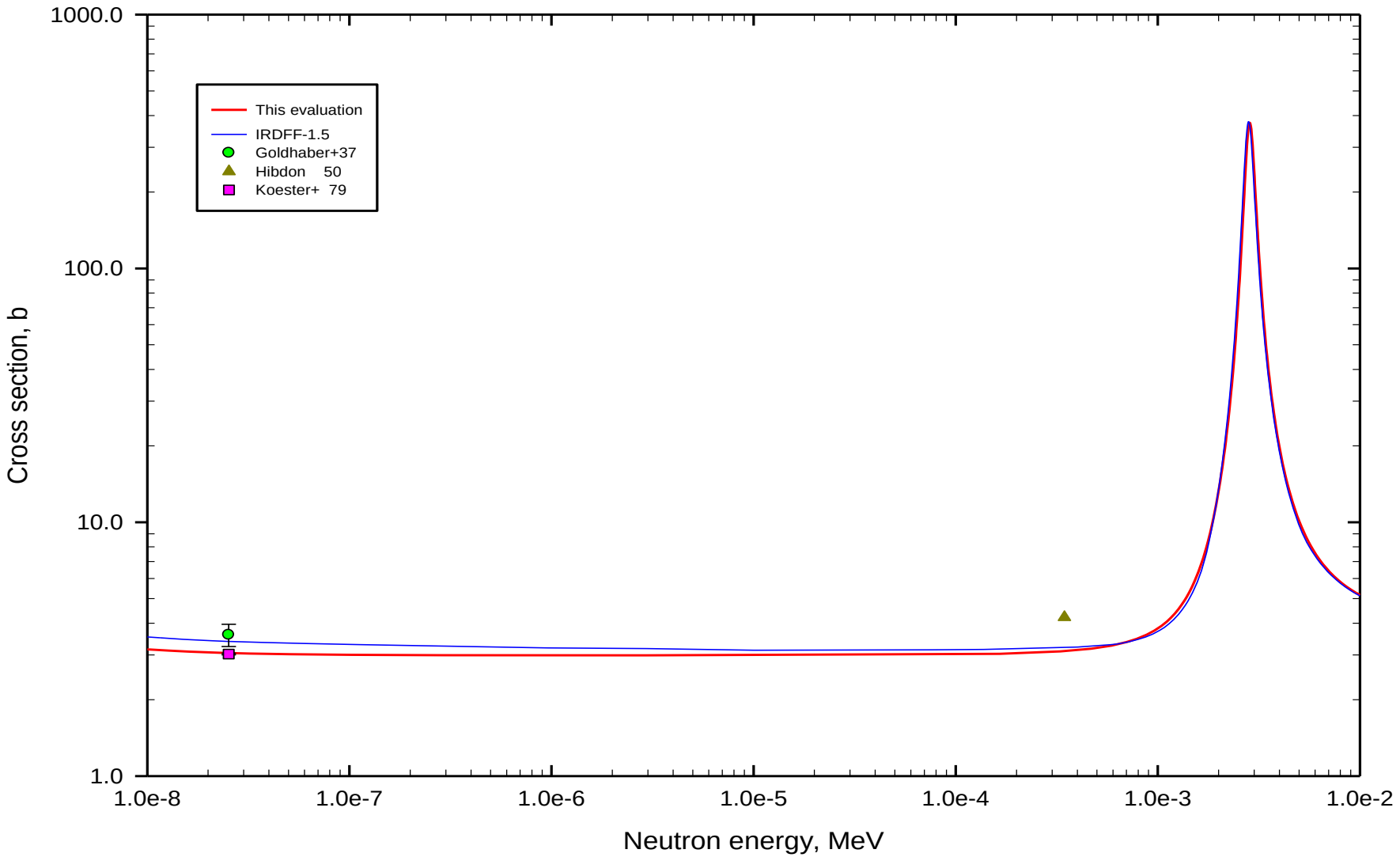

FIG. 3.1. Reconstructed from evaluated RM parameters ^{23}Na(n,el) excitation function in the energy range 1.0E-8 – 1.0E-2 MeV

in comparison with equivalent IRDFF-v1.5 and experimental data.

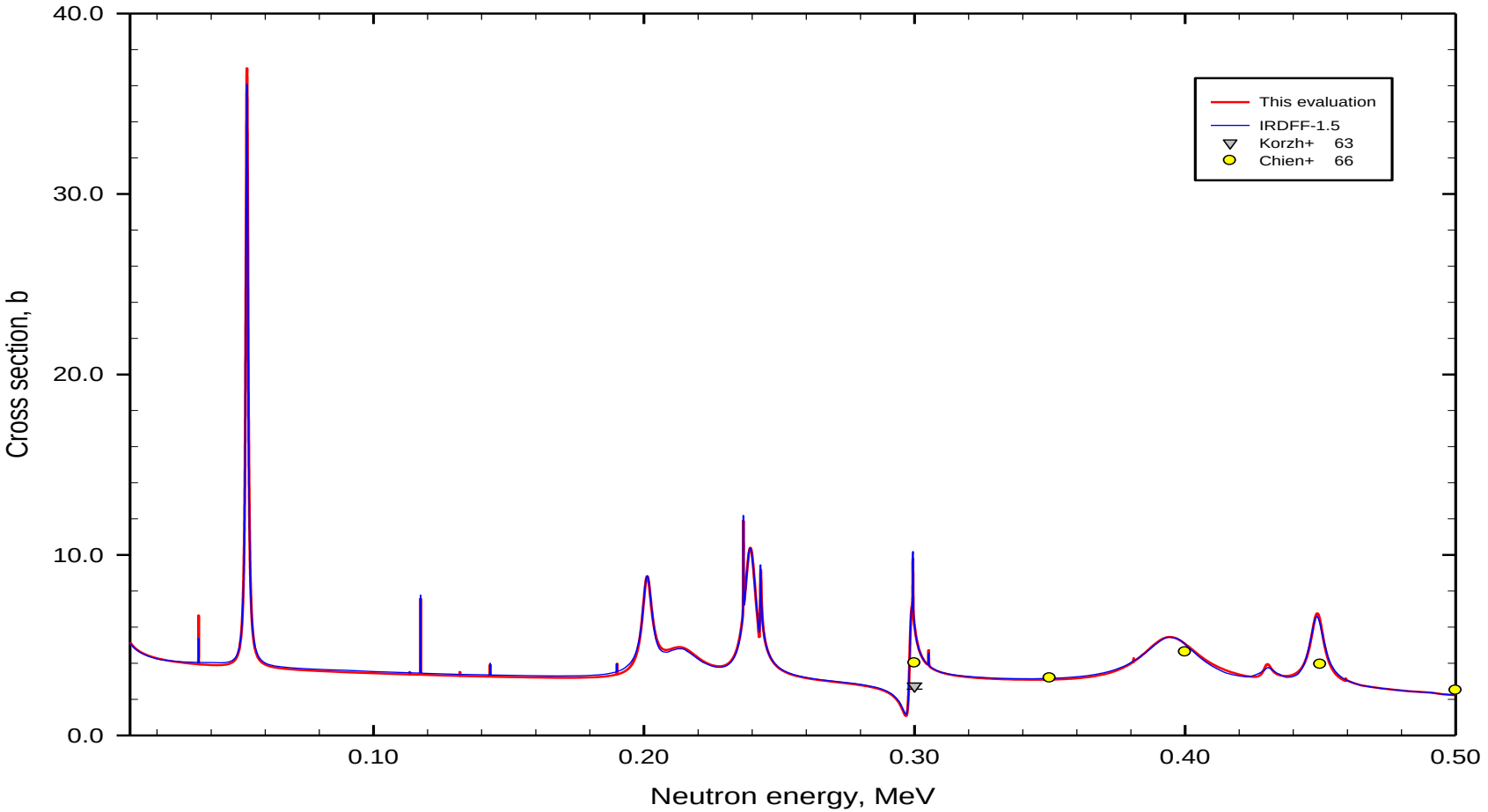

FIG. 3.2. Reconstructed from evaluated RM parameters ^{23}Na(n,el) excitation function in the energy range 0. 01 – 0.50 MeV in comparison with equivalent IRDFF-v1.5 and experimental data.

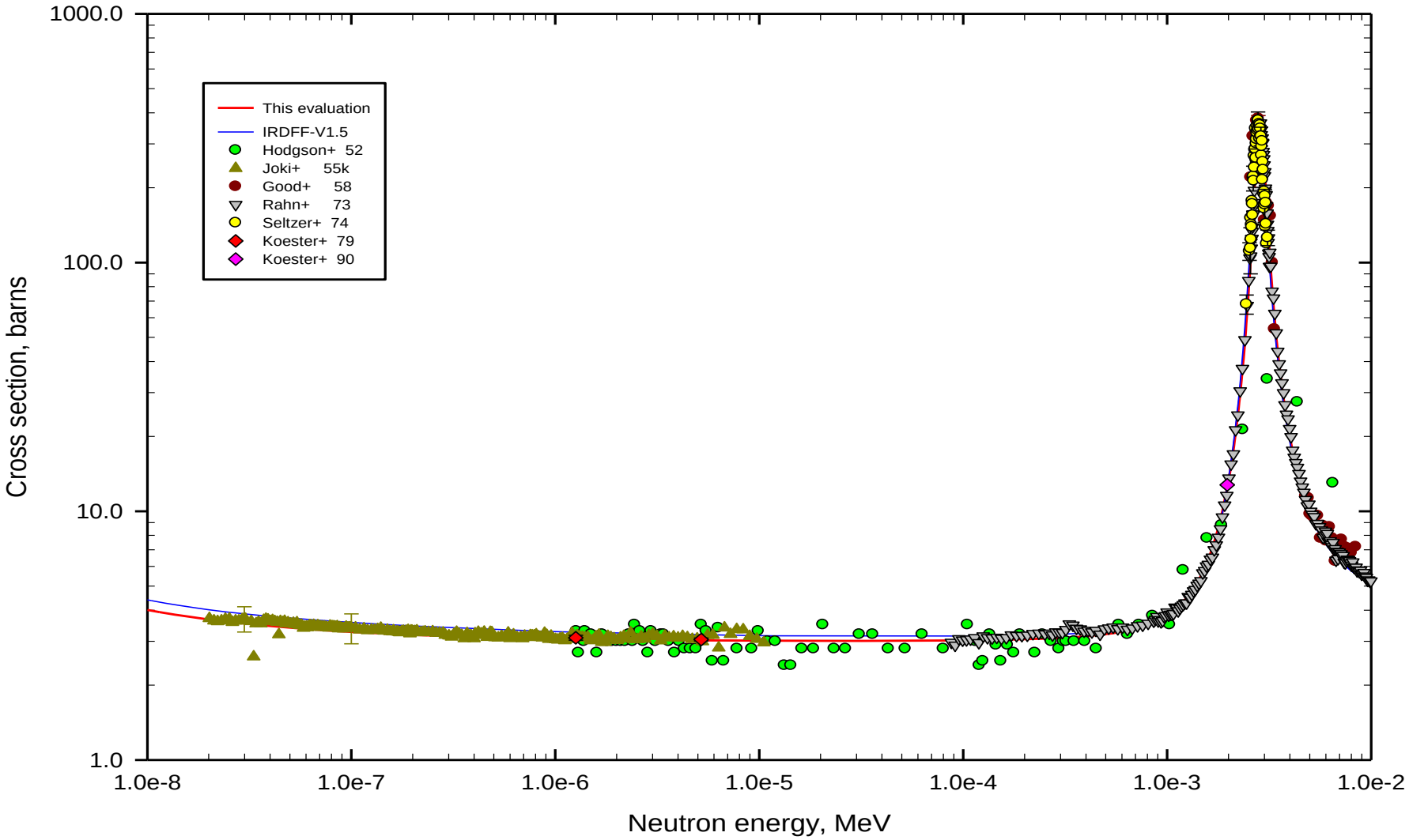

FIG. 3.3. Reconstructed from evaluated RM parameters ^{23}Na(n,tot) excitation function in the energy range 1.0E-8 – 1.0E-2 MeV in comparison with equivalent IRDFF-v1.5 and experimental data.

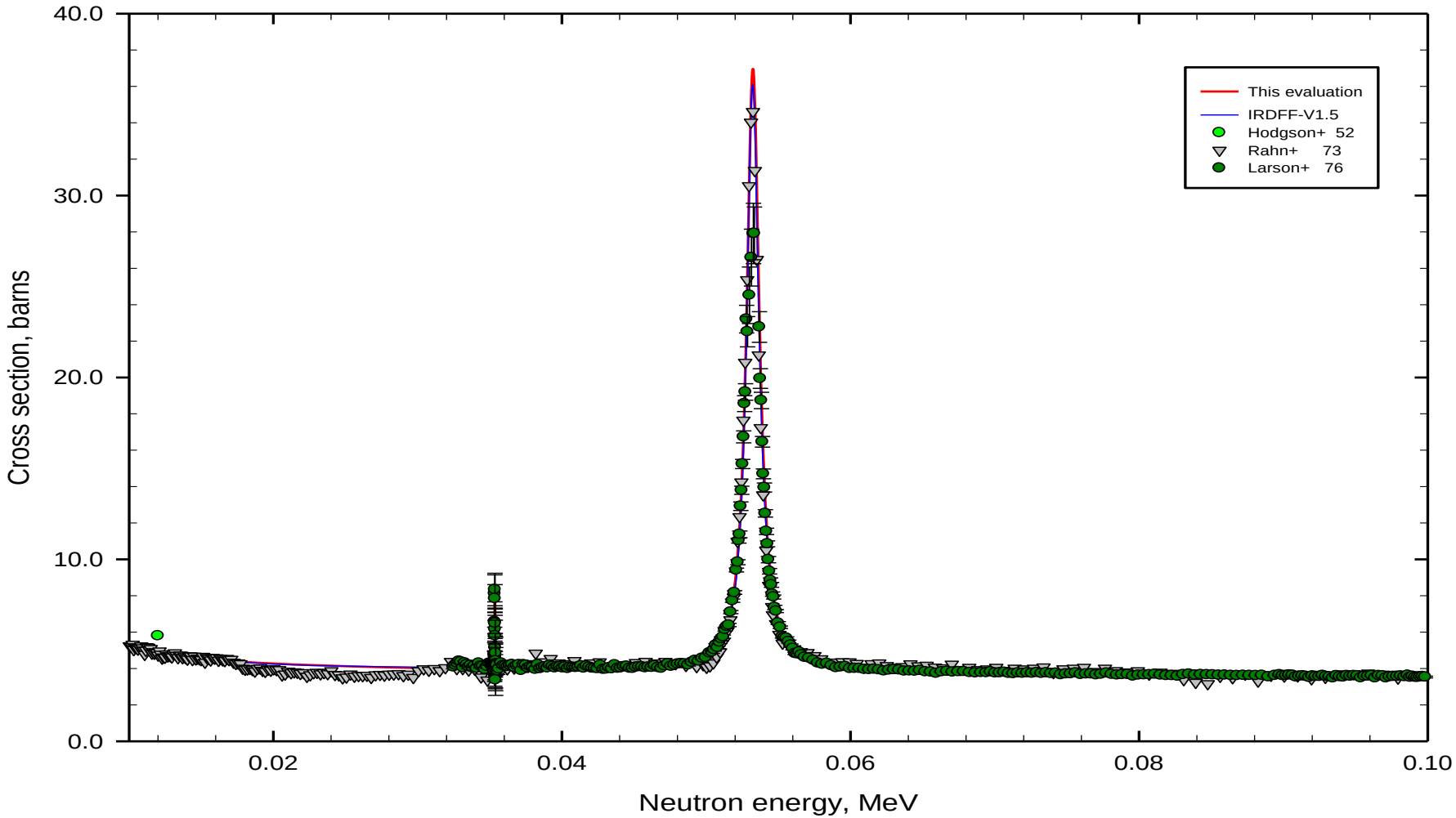

FIG. 3.4. Reconstructed from evaluated RM parameters ^{23}Na(n,tot) excitation function in the energy range 0.01 – 0.10 MeV in comparison with equivalent IRDFF-v1.5 and experimental data.

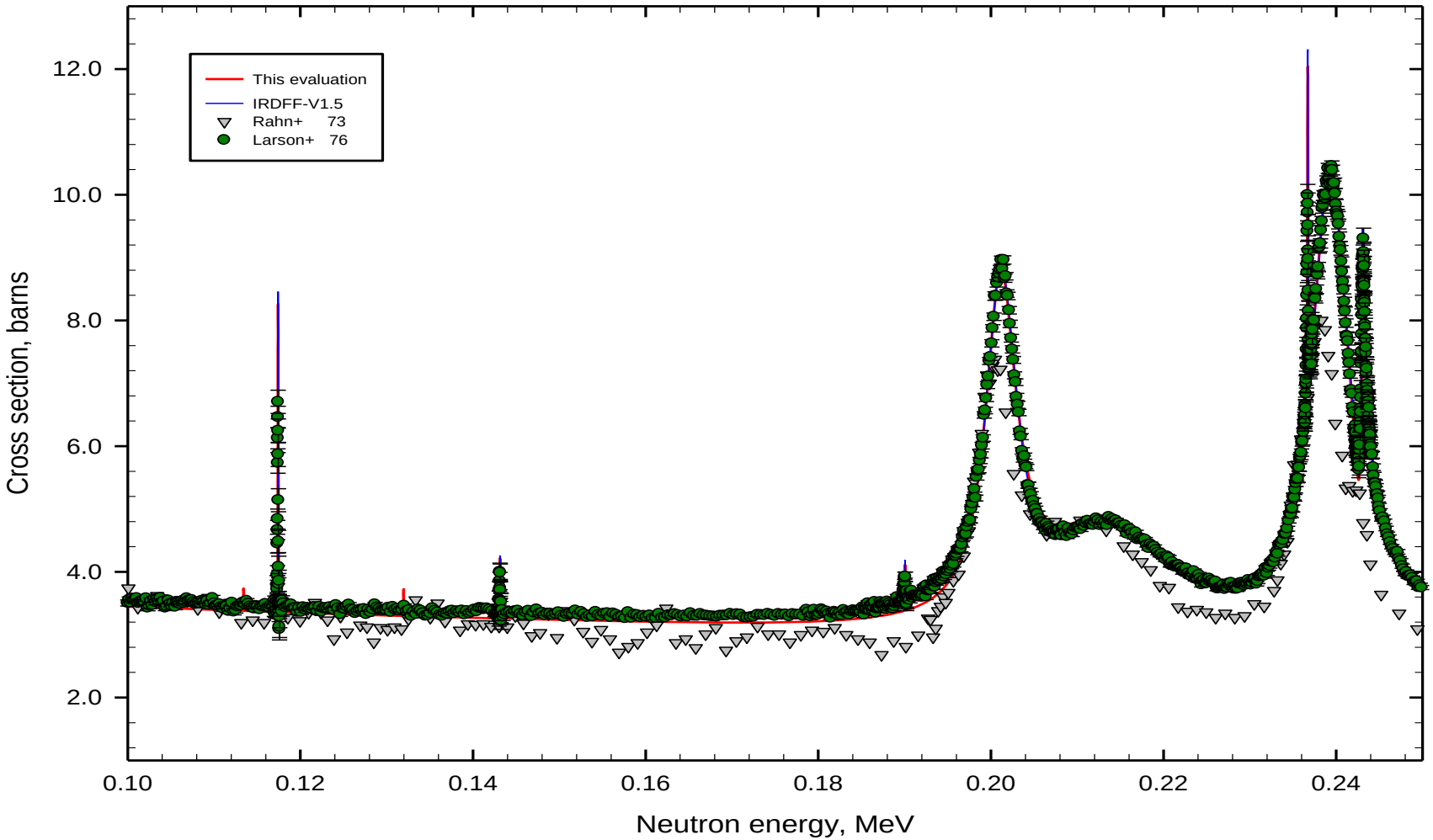

FIG. 3.5. Reconstructed from evaluated RM parameters ^{23}Na(n,tot) excitation function in the energy range 0.10 – 0.25 MeV in comparison with equivalent IRDFF-v1.5 and experimental data.

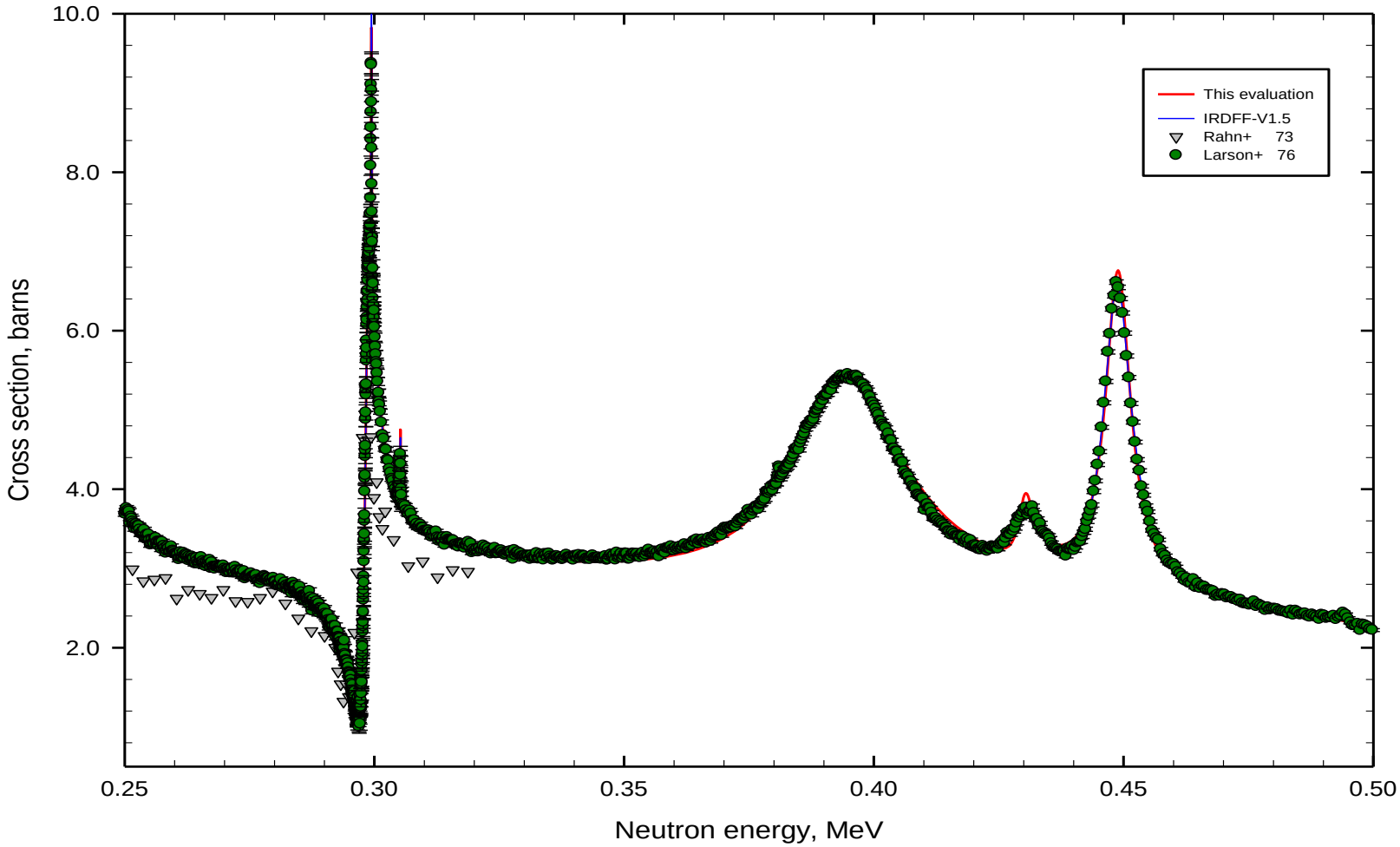

FIG. 3.6. Reconstructed from evaluated RM parameters ^{23}Na(n,tot) excitation function in the energy range 0.25 – 0.50 MeV in comparison with equivalent IRDFF-v1.5 and experimental data.

The excitation function for the $^{23}Na(n,\gamma)^{24}Na$ reaction in the energy region from 4.593E+5 eV to 20 MeV was re-evaluated by means of statistical analysis of experimental cross section data [3.15], [3.17-3.18], [3.27-3.29], [3.34] and [3.36-3.37]. Data obtained from the GNASH calculation were also included in the data base for evaluation. Evaluation of the $^{23}Na(n,\gamma)^{24}Na$ reaction cross section and related uncertainties was performed by means of the PADE-2 code.

Uncertainties in the re-evaluated $^{23}Na(n,\gamma)^{24}Na$ reaction excitation function are given via three independent matrixes.
In the RRR range 1.000E-5 - 7.000E+3 eV uncertainties are presented in the form of relative covariance matrix for the 37 neutron energy groups (LB=5). The covariance matrix was generated by the PADE-2 code. Eigenvalues of the 6-digit relative covariance matrix given in the 33-file are the following:

```
3.56851E-08     6.40933E-08     1.00164E-07     1.59356E-07
2.98557E-07     5.88895E-07     1.13833E-06     2.13791E-06
3.77277E-06     6.44677E-06     1.09760E-05     1.86934E-05
2.80614E-05     4.23139E-05     6.41327E-05     7.58830E-05
9.40970E-05     1.11013E-04     1.41451E-04     2.13710E-04
2.56772E-04     2.76387E-04     3.04995E-04     3.22681E-04
4.03195E-04     1.20835E-03     1.38475E-03     1.83188E-03
2.40534E-03     3.96628E-03     4.12075E-03     9.99763E-03
1.19163E-02     1.63392E-02     2.14604E-02     3.54159E-02
5.24765E-02
```

In the RRR range 7.000E+3 - 4.593E+5 eV uncertainties are given in the form of a diagonal matrix which includes data for the 13 neutron energy intervals (LB=1). Uncertainties in cross sections were calculated by means of the DSIGNG code [21] from uncertainties in Reich Moore parameters.
In the energy range 459.3 keV - 20 MeV uncertainties are presented in the form of a relative covariance matrix for the 39 neutron energy groups (LB=5). The covariance matrix was also generated by the PADE-2 code. Eigenvalues of the 6-digit relative covariance matrix given in the 33-file are the following:

```
6.10304E-09     8.32431E-09     9.98037E-09     1.37937E-08
1.43762E-08     1.47822E-08     1.60627E-08     1.70701E-08
2.05017E-08     2.13644E-08     2.25922E-08     2.39992E-08
2.53135E-08     2.56678E-08     2.61637E-08     2.72845E-08
2.86929E-08     2.94895E-08     2.98116E-08     3.07448E-08
3.09984E-08     3.76723E-08     1.06287E-06     1.07037E-04
7.99978E-04     1.02508E-03     1.51930E-03     3.11102E-03
5.20458E-03     7.92777E-03     9.59545E-03     1.05297E-02
1.33695E-02     1.54593E-02     1.76644E-02     1.90960E-02
2.45388E-02     3.59448E-02     4.72214E-02
```

Eigenvalues of the covariance sub-matrixes were tested, in addition, by means of the COVEIG code [3.71]. Some differences in the eigenvalues determined by the PADE-2 code and COVEIG code only occur for the lowest eigenvalues. The highest of the eigenvalues are equivalent.

Group cross sections and their uncertainties for the re-evaluated $^{23}Na(n,\gamma)^{24}Na$ reaction excitation function from 1.000E-05 eV to 20MeV are given in Table 3.3. Group boundaries are the same as in File-33

TABLE 3.3. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{23}Na(n,\gamma)^{24}Na$ REACTION IN THE NEUTRON ENERGY RANGE FROM 1.000E-5 EV TO 20 MEV.

Neutron energy (eV) from to	Cross section (barns)	Uncer-tainty (%)	Neutron energy (eV) from to	Cross section (barns)	Uncer-tainty (%)
1.000E-05 - 1.000E-04	1.282E+1	5.66	6.000E+04 - 8.000E+04	6. 694E-5	11.44
1.000E-04 - 1.000E-03	4.054E+0	5.44	8.000E+04 - 1.000E+05	2. 294E-5	6.18
1.000E-03 - 1.000E-02	1.282E+0	5.66	1.000E+05 - 2.000E+05	1.168E-3	22.53
1.000E-02 - 2.000E-02	6.989E-1	2.79	2.000E+05 - 3.000E+05	9.542E-4	22.23
2.000E-02 - 3.000E-02	5.363E-1	2.09	3.000E+05 - 4.593E+05	3.254E-4	16.18
3.000E-02 - 5.000E-02	4.252E-1	2.56	4.593E+05 - 5.000E+05	2.689E-4	8.32
5.000E-02 - 7.000E-02	3.457E-1	3.87	5.000E+05 - 5.500E+05	3.068E-4	7.29
7.000E-02 - 1.000E-01	2.905E-1	5.25	5.500E+05 - 6.000E+05	3.304E-4	7.46
1.000E-01 - 1.500E-01	2.399E-1	6.51	6.000E+05 - 6.500E+05	3.319E-4	8.02
1.500E-01 - 2.500E-01	1.902E-1	7.13	6.500E+05 - 7.000E+05	3.208E-4	7.96
2.500E-01 - 4.000E-01	1.490E-1	6.73	7.000E+05 - 7.500E+05	3.040E-4	7.66
4.000E-01 - 6.300E-01	1.184E-1	6.10	7.500E+05 - 8.000E+05	2.852E-4	7.59
6.300E-01 - 1.000E+00	9.412E-2	6.02	8.000E+05 - 9.000E+05	2.593E-4	7.20
1.000E+00 - 1.600E+00	7.456E-2	6.19	9.000E+05 - 1.000E+06	2.351E-4	5.68
1.600E+00 - 2.500E+00	5.937E-2	5.95	1.000E+06 - 1.250E+06	2.255E-4	7.67
2.500E+00 - 4.000E+00	4.722E-2	5.46	1.250E+06 - 1.500E+06	2.281E-4	6.52
4.000E+00 - 6.300E+00	3.754E-2	5.51	1.500E+06 - 1.750E+06	2.248E-4	6.01
6.300E+00 - 1.000E+01	2.991E-2	6.09	1.750E+06 - 2.000E+06	2.162E-4	6.12
1.000E+01 - 1.600E+01	2.376E-2	6.45	2.000E+06 - 2.500E+06	2.014E-4	6.21
1.600E+01 - 2.000E+01	2.015E-2	6.38	2.500E+06 - 3.000E+06	1.842E-4	6.24
2.000E+01 - 3.000E+01	1.724E-2	6.11	3.000E+06 - 3.500E+06	1.712E-4	6.01
3.000E+01 - 4.000E+01	1.463E-2	5.91	3.500E+06 - 4.000E+06	1.620E-4	5.66
4.000E+01 - 6.000E+01	1.239E-2	5.94	4.000E+06 - 4.500E+06	1.561E-4	5.36
6.000E+01 - 8.000E+01	1.059E-2	6.11	4.500E+06 - 5.000E+06	1.537E-4	5.19
8.000E+01 - 1.000E+02	9.457E-3	6.20	5.000E+06 - 5.500E+06	1.555E-4	5.25
1.000E+02 - 2.000E+02	7.721E-3	6.02	5.500E+06 - 6.000E+06	1.634E-4	5.58
2.000E+02 - 3.000E+02	6.366E-3	5.62	6.000E+06 - 6.500E+06	1.775E-4	5.61
3.000E+02 - 4.000E+02	5.782E-3	5.47	6.500E+06 - 7.000E+06	1.866E-4	5.52
4.000E+02 - 6.000E+02	5.452E-3	5.44	7.000E+06 - 7.500E+06	1.793E-4	6.00
6.000E+02 - 8.000E+02	5.453E-3	5.40	7.500E+06 - 8.000E+06	1.668E-4	7.14
8.000E+02 - 1.000E+03	5.801E-3	5.23	8.000E+06 - 8.500E+06	1.588E-4	6.88
1.000E+03 - 2.000E+03	1.024E-2	4.27	8.500E+06 - 9.000E+06	1.552E-4	6.28
2.000E+03 - 3.000E+03	1.336E-1	4.20	9.000E+06 - 1.000E+07	1.556E-4	5.94
3.000E+03 - 4.000E+03	4.225E-2	4.44	1.000E+07 - 1.100E+07	1.617E-4	6.04
4.000E+03 - 5.000E+03	3.876E-3	5.07	1.100E+07 - 1.200E+07	1.726E-4	5.98
5.000E+03 - 6.000E+03	1.337E-3	16.78	1.200E+07 - 1.300E+07	1.869E-4	5.86
6.000E+03 - 7.000E+03	6.703E-4	19.24	1.300E+07 - 1.400E+07	2.026E-4	6.04
7.000E+03 - 8.000E+03	4.155E-3	6.41	1.400E+07 - 1.450E+07	2.133E-4	6.34
8.000E+03 - 1.000E+04	2.395E-4	5.80	1.450E+07 - 1.500E+07	2.182E-4	6.42
1.000E+04 - 1.600E+04	1.074E-4	4.31	1.500E+07 - 1.600E+07	2.190E-4	6.26
1.600E+04 - 2.000E+04	6.063E-5	2.98	1.600E+07 - 1.700E+07	2.045E-4	6.91
2.000E+04 - 3.000E+04	4.549E-5	3.18	1.700E+07 - 1.800E+07	1.705E-4	9.22
3.000E+04 - 4.000E+04	4.180E-3	18.79	1.800E+07 - 1.900E+07	1.262E-4	11.12
4.000E+04 - 6.000E+04	2.431E-3	21.65	1.900E+07 - 2.000E+07	8.427E-5	19.97

Uncertainties in the evaluated $^{23}Na(n,\gamma)^{24}Na$ excitation function range from 2.09% to 22.53%. The smallest uncertainty of 2.09% in the evaluated cross sections is observed in the neutron energy interval 2.000E-02 - 3.000E-02 eV and reflect the knowledge of 2200 m/sec capture cross section. Uncertainties in the $^{23}Na(n,\gamma)^{24}Na$ reaction cross sections exceeding 10% are observed in the resonance range between 5.000E+03 - 7.000E+03 eV, 3.000E+04 - 6.000E+04 eV, 6.000E+04 - 8.000E+04 eV, 1.000E+05 - 4.593E+05 eV and in the energy range 18 - 20 MeV.

The re-evaluated excitation function for the $^{23}Na(n,\gamma)^{24}Na$ reaction in the neutron energy range 1.000e-10 - 20 MeV is shown in Fig. 3.7 in comparison with the equivalent data from the IRDFF-v1.5 library and corrected experimental data. The same information, but in the narrow neutron energy range from threshold to 1.000E-1 - 20 MeV, is shown in Fig. 3.8.

As can be seen from Fig. 3.7 the re-evaluated the $^{23}Na(n,\gamma)^{24}Na$ reaction excitation function and data from IRDFF-v1.5 library below 0.7 keV are practically the same. Significant differences in the behavior of the compared evaluations occur above the first 2.8465 keV resonance. This might be explained by differences in the used resonance parameters. In the IRDFF-v1.5 library single level Breit-Wigner parameters were used.

Integral cross sections of the $^{23}Na(n,\gamma)^{24}Na$ reaction have been measured by means of the activation method in three different neutron fields.
Hughes et al. measured the integral cross section in the ^{235}U thermal-neutron induced fission spectrum [3.76-3.77] by means of the activation method. Fission neutrons were generated at the thermal beam of the ANL reactor. The $^{23}Na(n,\gamma)^{24}Na$ integral cross-section obtained in this work is equal to $\langle\sigma\rangle_{U\text{-}235}$ = (0.280 ± 0.019) mb. The original experimental value was corrected for the $^{23}Na(n,\gamma)^{24}Na$ thermal cross-section of (530.4±11.2) mb, with Fc= 0.96436.

The second integral experiment was performed by Z. Dezso and J. Csikai with ^{252}Cf source at the Institute of Experimental Physics, Kossuth University, Debrecen, Hungary [3.78]. Measurements with spontaneous fission neutron spectrum of ^{252}Cf were carried out by means of the activation method in scattering free arrangement. Neutron flux was determined by two activation detectors with using reactions $^{27}Al(n,\alpha)^{24}Na$ and $^{115}In(n,n')^{115m}In$. The Gamma spectrum of irradiated Na, Al and In samples was measured with the Ge(Li) detector. The $^{23}Na(n,\gamma)^{24}Na$ integral cross-section obtained in this work is equal to $\langle\sigma\rangle_{Cf\text{-}252}$ = (0.316 ± 0.014) mb. The original experimental value was corrected to the new standards for monitor reactions, Fc= 0.9444.

The third integral experiment was performed by G.P. Lamaze et al. from NBS USA at the Intermediate-Energy Standard Neutron Field (ISNF) [3.77]. The ISNF spectrum resembles the fast reactor spectrum, but has a less complex structure. Median energy of the spectrum is equal to 0.58 MeV. The 98% of the neutron fluence is between 1.2 keV and 5.6 MeV. Measurements in the ISNF spectrum were carried out by means of the activation method. Na samples were prepared in the form of reagent grade salt pellet, 12.7mm diameter, 1.3mm thick. Neutron fluence was determined from the $^{252}Cf/^{235}U$ count ratio for ^{140}Ba-^{140}La chain. The measured count rate was equal to 0.772.
Gamma spectra of irradiated samples were measured by Ge(IN) detector. The $^{23}Na(n,\gamma)^{24}Na$ integral cross-section measured in the ISNF spectrum is equal to $\langle\sigma\rangle$ = (1.57 ± 0. 10) mb.

The re-evaluated excitation function for the $^{23}Na(n,\gamma)^{24}Na$ 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 ISNF spectrum are compared with the IRDFF-v1.5 and experimental data in Table 3.4.

TABLE 3.4. CALCULATED AND MEASURED AVERAGED CROSS SECTIONS FOR THE $^{23}Na(n,\gamma)^{24}Na$ REACTION IN A THREE NEUTRON SPECTRA.

Type of neutron field	Averaged cross section, μb		90% response function, MeV	*C/E*
	Calculated	Measured		
^{235}U thermal fission neutron spectrum	0.27546 [A] 0.27109 [B]	0.280 ± 0.019 [3.76]	5.25E-2 - 4.30E+0 5.25E-2 - 4.40E+0	0.98379 0.96818
^{252}Cf spontaneous fission neutron spectrum	0.27621 [A] 0.27162 [B]	0.316 ± 0.014 [3.78]	5.25E-2 - 4.60E+0 5.25E-2 - 4.70E+0	0.87408 0.85956
ISNF neutron spectrum	1.8891 [A] 1.9165 [B]	1.57 ± 0.10 [3.79]	1.70E-3 - 7.20E-1 1.70E-3 - 7.20E-1	1.20325 1.22070

[A] - Present re-evaluation,
[B] - IRDFF-v1.5

The 90% response function shows the neutron energy range where the investigated excitation function is tested in the benchmark spectrum. The value of C/E given in the last column is the ratio of the calculated to experimental cross sections.

Integral cross sections for ^{235}U thermal fission neutron spectrum calculated from the re-evaluated excitation function and IRDFF-v1.5 data agree with experimental data [3.72] in the limit of its uncertainty. At the same time calculated and measured integral cross sections for ^{252}Cf spontaneous fission neutron spectrum and ISNF spectrum differ significantly. It is difficult to explain precisely the reason for such a big contradiction between the data. It is evident that experimental data by Z. Dezso and J. Csikai [3.78] are overestimated. It may also be supposed that experimental data for ISNF spectrum by G.P. Lamaze et al. [3.79] are underestimated.

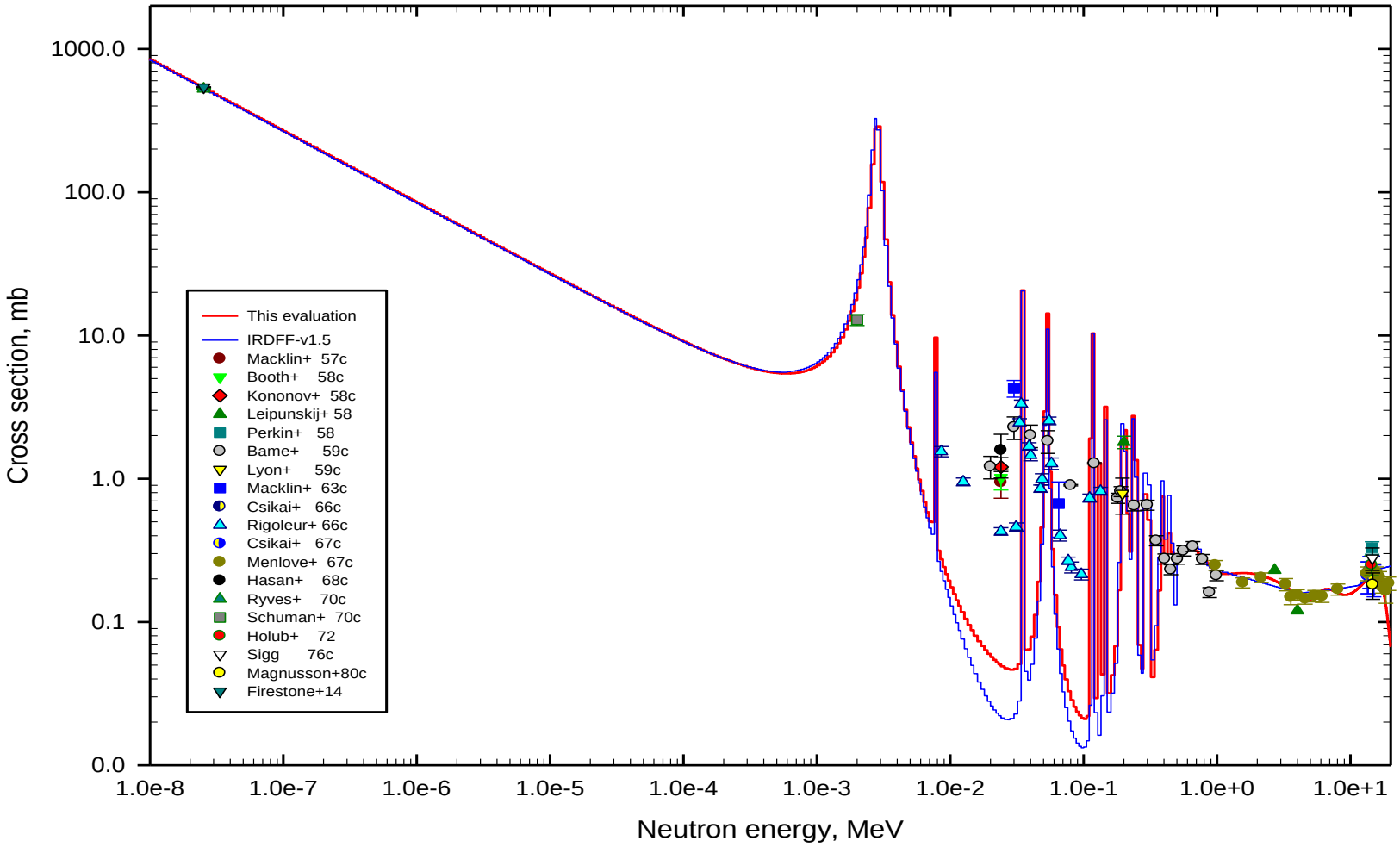

FIG. 3.7. Re-evaluated $^{23}Na(n,\gamma)^{24}Na$ reaction excitation function in the energy range 1.0E-8 – 20 MeV in comparison with equivalent data from IRDFF-v1.5 library and corrected experimental data.

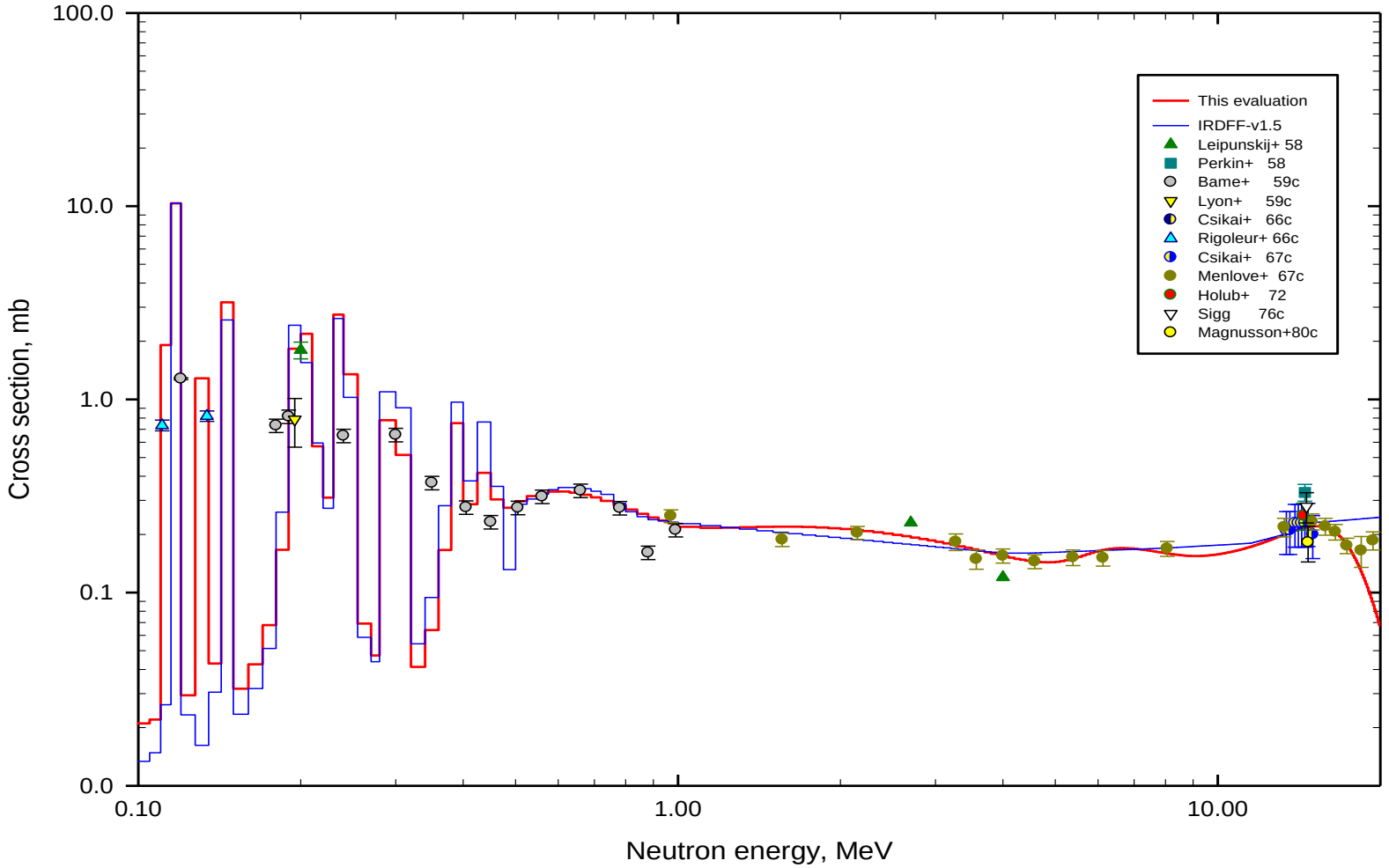

FIG. 3.8. Re-evaluated $^{23}Na(n,\gamma)^{24}Na$ reaction excitation function in the energy range 0.10 – 20 MeV in comparison with equivalent data from IRDFF-v1.5 library and corrected experimental data.

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4. RE-EVALUATION OF THE ^{23}Na(n,2n)^{22}Na REACTION EXCITATION FUNCTION

The (J_π=3+) ground level of ^{22}Na obtained in the (n,2n) reaction undergoes 100% decay via β^+ decay mode with a half-life of (2.6018 ± 0.0022) years. The β^+ decay is accompanied by emission of X-ray and gamma-ray radiation. In the gamma-ray spectrum the annihilation 511-keV line I_γ = (1.8076 ± 0.0004) and 1274.537-keV line I_γ = (0.99940 ± 0.00014) are observed. Recommended decay data for the half-life and gamma ray emission probability per decay of ^{22}Na were taken from Ref. [4.21].

Microscopic experimental data for the ^{23}Na(n,2n)^{22}Na reaction excitation function are given in the works [4.1- 4.20] and cover the neutron energies range from 12.63 MeV to 66.0 MeV. Experimental data of works [4.1-4.2], [4.4-4.8], [4.10-4.11], [4.14-4.15] and [4.18-4.19] 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 ^{22}Na.

Special correction was applied to the experimental data [4.2], [4.3], [4.10], [4.12], [4.15] and [4.17].
Corrected to the new standards experimental data of Liskien and Paulsen [4.2] were renormalized to the integral of cross section in the energy range 13.50-19.45 MeV calculated from representative experimental data [4.6], [4.9], [4.11], [4.13], [4.16], [4.18] and [4.19], Fc= 0.57843.
Original experimental data by Picard and Williamson [4.3] in the energy range 14.89-21.00 MeV were renormalized at 19.61 MeV to a value of 115.5 mb, determined from experimental data of Strohmaier et al. (E_n = 19.449 MeV, $\sigma_{n.2n}$ = (114.5±6.6) mb) [4.13], Fc= 1.81604.
Corrected to the new standards experimental data of Adamski et al. [4.10] were renormalized for a preliminary evaluated cross section value of 55.94 mb (± 2.08%) at 15.50 MeV, Fc= 1.17942.
Measured by Xu Zhizheng et al. [4.12] cross sections in the energy interval 13.30-18.50 MeV were renormalized to a preliminary evaluated value of 41.76 mb (±1.72%) at 14.90 MeV, Fc= 0.59657.
Cross sections measured by Lu Hanlin et al. at the six energy points between 13.50-17.98 MeV [4.15] after applied correction to a new standards were renormalized to the experimental data of Filatenkov and Chuvaev [19] in the overlapping energy interval 13.56 - 14.78 MeV, Fc= 0.88720.
Using activation method and unfolding procedure Uwamino et al. measured the ^{23}Na(n,2n)^{22}Na reaction excitation function in a wide energy diapason 13.50 – 38.50 MeV [4.17]. These experimental data above 20 MeV were renormalized to a preliminary evaluated cross section of 121.56 mb (±4.87%) at 21.50 MeV, Fc=1.61370.

The excitation function for the ^{23}Na(n,2n)^{22}Na reaction in the energy region from threshold to 60 MeV was evaluated by means of statistical analysis of experimental cross section data [4.1-4.3], [4.5-4.6], [4.8-4.15] and [4.17-4.20].

The evaluation of the ^{23}Na(n,2n)^{22}Na reaction cross section and related uncertainties was performed by means of the PADE-2 code. Uncertainties in the evaluated excitation function for the ^{23}Ni(n,2n)^{22}Na reaction are given in the form of a relative covariance matrix for 44-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 occur at the lowest eigenvalues. The highest of the eigenvalues are equivalent. Six-digit eigenvalues calculated by the PADE-2 code for the relative covariance matrix in File-33 are as follows:

```
1.29058E-08     1.35549E-08     1.40973E-08     1.52887E-08
1.66625E-08     1.77036E-08     1.87318E-08     2.05388E-08
2.23059E-08     2.45214E-08     2.73970E-08     2.98716E-08
3.41843E-08     3.76746E-08     4.70351E-08     5.28023E-08
7.33572E-08     8.42869E-08     1.12040E-07     1.36473E-07
1.60603E-07     1.98813E-07     2.20865E-07     2.80909E-07
3.28825E-07     3.66408E-07     4.50932E-07     5.46932E-07
6.14939E-07     6.68059E-07     7.80246E-07     9.10478E-07
1.03278E-06     1.36324E-06     3.83598E-06     2.43399E-05
1.03707E-03     1.69749E-03     2.98424E-03     3.22162E-03
6.70516E-03     5.17577E-02     5.69772E-02     4.35821E-01
```

Group cross sections and their uncertainties for the re-evaluated $^{23}Na(n,2n)^{22}Na$ reaction excitation function 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 $^{23}Na(n,2n)^{22}Na$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 60 MeV.

Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)	Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)
12.965 - 13.400	1.453	6.87	23.000 - 24.000	129.175	7.59
13.400 - 13.600	4.877	2.89	24.000 - 25.000	123.233	8.41
13.600 - 13.800	8.245	2.33	25.000 - 26.000	114.985	8.71
13.800 - 14.000	12.508	2.24	26.000 - 27.000	105.609	8.60
14.000 - 14.200	17.552	2. 01	27.000 - 28.000	96.037	8.27
14.200 - 14.400	23.195	1.74	28.000 - 29.000	86.875	7.91
14.400 - 14.600	29.184	1.60	29.000 - 30.000	78.449	7.65
14.600 - 14.800	35.264	1.60	30.000 - 32.000	67.543	7.56
14.800 - 15.000	41.214	1.67	32.000 - 34.000	55.766	7.96
15.000 - 15.500	50.789	1.76	34.000 - 36.000	46.758	8.67
15.500 - 16.000	62.490	1.91	36.000 - 38.000	39.848	9.48
16.000 - 16.500	71.998	2.08	38.000 - 40.000	34.485	10.38
16.500 - 17.000	80.057	2.17	40.000 - 42.000	30.261	11.44
17.000 - 17.500	87.249	2.17	42.000 - 44.000	26.883	12.73
17.500 - 18.000	93.945	2.11	44.000 - 46.000	24.140	14.29
18.000 - 18.500	100.320	2.06	46.000 - 48.000	21.882	16.14
18.500 - 19.000	106.406	2.07	48.000 - 50.000	20.000	18.26
19.000 - 19.500	112.146	2.19	50.000 - 52.000	18.413	20.61
19.500 - 20.000	117.420	2.47	52.000 - 54.000	17.061	23.15
20.000 - 21.000	124.025	3.22	54.000 - 56.000	15.898	25.86
21.000 - 22.000	129.918	4.68	56.000 - 58.000	14.889	28.69
22.000 - 23.000	131.648	6.27	58.000 - 60.000	14.008	31.63

Uncertainties in the evaluated the $^{23}Na(n,2n)^{22}Na$ excitation function range from 1.60% to 31.63%. The smallest uncertainties in the evaluated cross sections 1.60-1.91% are observed in the neutron energy range 14.2 - 16.0 MeV. Uncertainties in the $^{23}Ni(n,2n)^{22}Na$ reaction cross sections exceeding 5% are observed near threshold and in the energy range 22 - 60 MeV.

The re-evaluated excitation function for the $^{23}Na(n,2n)^{22}Na$ reaction in the neutron energy range from threshold to 60 MeV is shown in Fig. 4.1 in comparison with the equivalent data from IRDFF-v1.5 library and corrected experimental data. The same information, but in the narrow neutron energy range from threshold to 20 MeV is shown in Fig. 4.2. The maximum of $^{23}Na(n,2n)^{22}Na$ reaction excitation function lies between 23-24.5 MeV, because of the competition channel – reaction $^{23}Na(n,3n)^{21}Na$ having a threshold equal to 24.518 MeV. The next competition channel – reaction $^{23}Na(n,4n)^{20}Na$ have a threshold equal to 42.375 MeV.

Integral cross sections of the $^{23}Na(n,2n)^{22}Na$ reaction have been measured by means of the activation method in three different reactors [4.23], [4.25], [4.26] and critical assembly [4.24]. In the results obtained measured integral cross sections range from 1.200 micro barn to (4.30 ± 0.80) micro barn (μb).

The lowest value of 1.200 μb was obtained by K. Wagner in measurements at the Rossendorf reactor [4.23]. Information about uncertainty in cross section and neutron flux determination is not given in the EXFOR library. The maximal value of (4.30 ± 0.80) μb has been measured by M. Martini, P. Moioli and C. Romanelli at the fast reactor TAPIRO [4.26]. Measurements were carried out at a fast spectrum neutron beam from the reactor. Uncertainty given for measured cross section is the statistical uncertainty.

The third experiment was performed by E. Steinnes at the JEEP I reactor (Kjeller, Norway) [4.25]. Irradiation of Na_2O_3 samples was carried out in the centre of the isotopic channel of the reactor. The author of this experiment proposed that neutron flux distribution above 2 MeV in the centre of the channel is similar to a pure fission spectrum. The neutron spectrum was determined by an activation detector using the $^{58}Ni(n,p)^{58}Co$ reaction. The activity of the irradiated samples was measured by a gamma spectrometer with a NaI(Tl) detector. Radiochemical separation was used after irradiation. After correction to the new standards the data of [4.25] give a value of $\langle\sigma\rangle_{U\text{-}235}$ = (2.042 ± 0.147) μb for the $^{23}Ni(n,2n)^{22}Na$ reaction.

F. Nasyrov and B.D. Sciborskij measured the $^{23}Na(n,2n)^{22}Na$ integral cross-section in an uranium critical assembly at the Kurchatov Institute, Moscow, Russian Federation [4.24]. The averaged cross section for ^{235}U thermal fission neutron spectrum was recalculated from a value measured in the core of critical assembly. The $^{23}Ni(n,2n)^{22}Na$ integral cross-section obtained in this work is equal to $\langle\sigma\rangle_{U\text{-}235}$ = (2.724 ± 0.706) μb. The original experimental value was corrected for new recommended decay data of ^{22}Na [4.21].

The integral experiment with ^{252}Cf source was performed by M. Kostal at the Research Centre Rez near Prague, Czech Republic [4.27]. Measurements with spontaneous fission neutron spectrum of ^{252}Cf were carried out by means of the activation method in scattering free arrangement. The neutron flux was calculated by means of the MCNPX 2.6.0 code with knowledge of absolute emission. The model covers details of all structural components. Absolute emission of the ^{252}Cf source was measured by manganese bath in NPL with an uncertainty of 0.9%. The gamma spectrum of irradiated Na samples has been measured by the HP Ge detector. The $^{23}Na(n,2n)^{22}Na$ integral cross-section obtained in this work is equal to $\langle\sigma\rangle_{Cf\text{-}252}$ = (8.697 ± 0.350) μb.

The re-evaluated excitation function for the $^{23}Na(n,2n)^{22}Na$ reaction was tested against the integral experimental data [4.24] and [4.27]. The calculated averaged cross sections over the ^{235}U thermal and ^{252}Cf spontaneous fission neutron spectra are compared with the IRDFF-v1.5 and experimental data in Table 4.2.

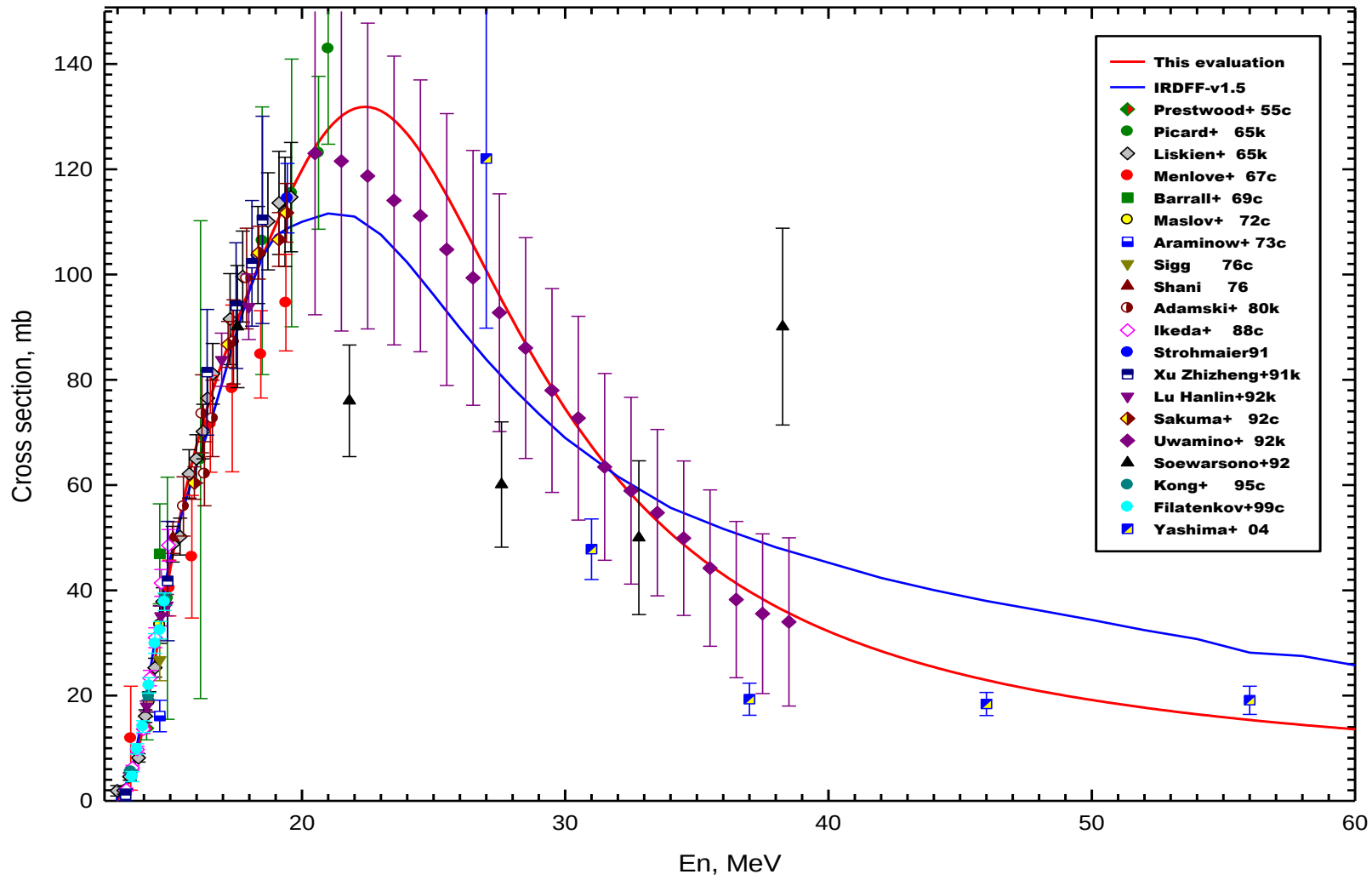

FIG. 4.1. Evaluated $^{23}Na(n,2n)^{22}Na$ reaction excitation function in the energy range from threshold to 60 MeV in comparison with equivalent data from IRDFF-v1.5 library and corrected experimental data.

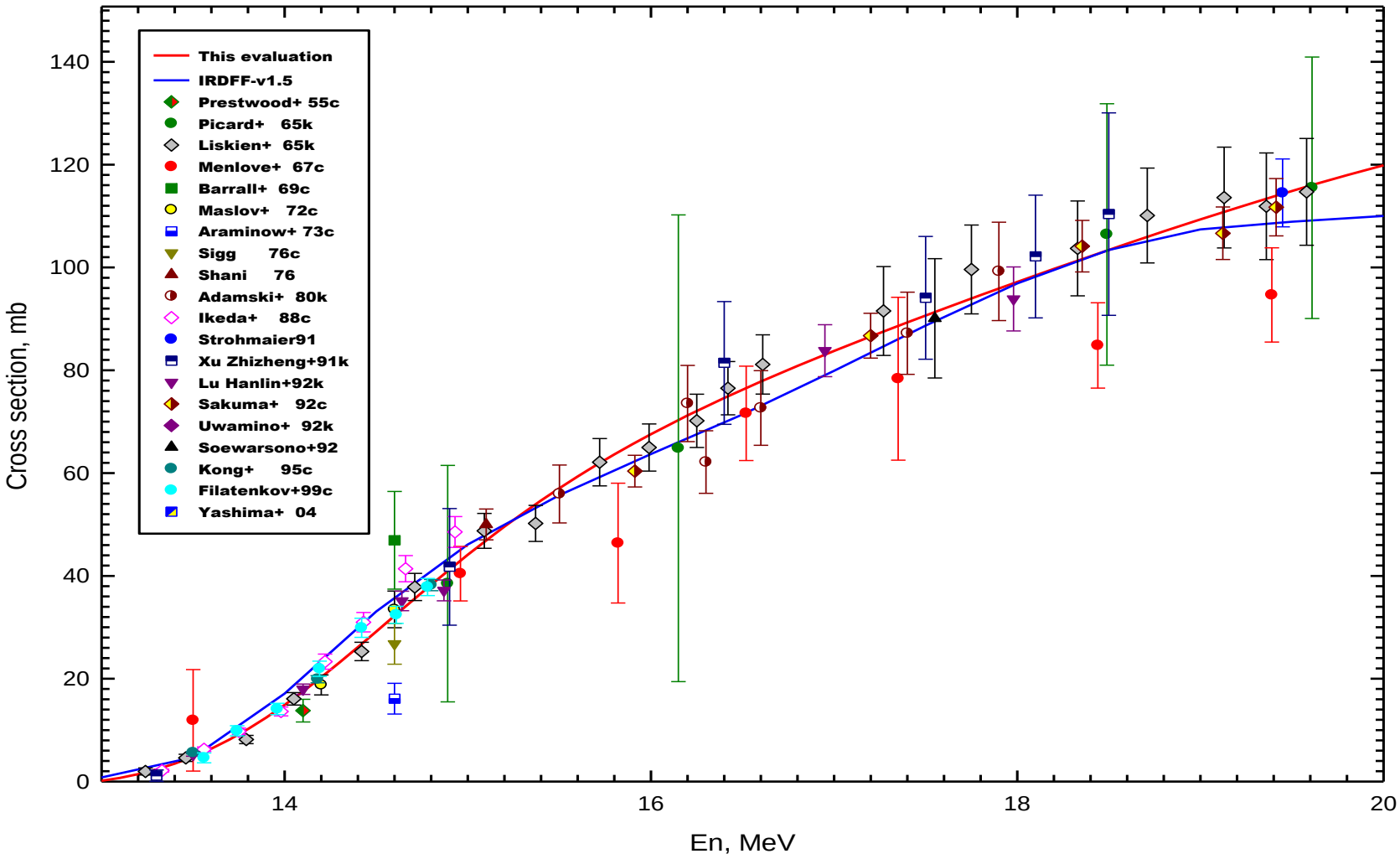

FIG. 4.2. Evaluated $^{23}Na(n,2n)^{22}Na$ reaction excitation function in the energy range from threshold to 20 MeV in comparison with equivalent data from IRDFF-v1.5 library and corrected experimental data.

TABLE 4.2. CALCULATED AND MEASURED AVERAGED CROSS SECTIONS FOR THE ^{23}Na(n,2n)^{22}Na REACTION IN ^{235}U THERMAL FISSION AND ^{252}Cf SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	Averaged cross section, μb		90% response function, MeV	*C/E*
	Calculated	Measured		
^{235}U thermal fission neutron spectrum	3.1373 [A] 3.2469 [B]	2.724 ± 0.706 [4.24]	13.70 - 18.80 13.60 - 18.60	1.15173 1.19196
^{252}Cf spontaneous fission neutron spectrum	8.6978 [A] 8.8865 [B]	8.697 ± 0.350 [4.27]	13.80 - 19.50 13.70 - 19.30	1.00009 1.02179

[A] - Present evaluation
[B] - IRDFF-v1.5

The 90% response function shows the neutron energies range where the investigated excitation function is tested in the benchmark spectrum.

The last column gives the value of C/E, which is a ratio of the calculated to experimental cross sections.

The C/E values obtained for the ^{235}U thermal fission neutron spectrum and ^{252}Cf spontaneous fission neutron spectrum show that the ^{23}Na(n,2n)^{22}Na integral cross sections calculated from both evaluations compared agree with the experimental value within its uncertainty. The $\langle\sigma\rangle_{U\text{-}235}$ and $\langle\sigma\rangle_{Cf\text{-}252}$ cross sections calculated from the RRDFF-v1.5 excitation function agree just a little worse with the relevant experimental value.

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5. EVALUATION OF THE $^{27}Al(n,2n)^{26}Al$ REACTION EXCITATION FUNCTION

The abundance of ^{27}Al isotope in natural aluminum is 100 atom percent. The 228.305-keV metastable level (J_π=0+) and ground level (J_π=5+) of ^{26}Al obtained in the (n,2n) reaction undergoes 100% via ε capture decay mode with a half-life of (6.3452 ± 0.0019) seconds and (717000 ± 24000) years, respectively. The ε capture decay in both cases is accompanied by emission of X-ray and gamma-ray radiation. The 511-keV annihilation gamma radiation (I_γ = 2.0) is normally used to determine the $^{27}Al(n,2n)^{26m}Al$ reaction rate. The most intensive lines in the gamma-ray spectrum accompanying ^{26g}Al decay are annihilation 511-keV line I_γ = (1.635 ± 0.004) and 1808.65-keV line I_γ = (0.9976 ± 0.0004). Recommended decay data for the half-life and gamma ray emission probability per decay of ^{26m}Al and ^{26g}Al were taken from Ref. [5.1].

Microscopic experimental data for the $^{27}Al(n,2n)^{26m}Al$ reaction excitation function are given in the works [5.2-5.7] and cover the neutron energy range from 13.94 MeV to 20.2 MeV. Experimental data of works [5.4], [5.5], [5.7] were, in the process of analysis, corrected to the new standards for the relevant monitor reactions (see Table 2.1) and to the recommended decay data for ^{26m}Al [5.1].

Special correction was applied to the experimental data of G.S. Mani, G.J. McCallum, A.T.G. Ferguson [5.2]. These experimental data were renormalized to the results of the corrected measurements of D.M. Arnold and L.A. Rayburn [5.4] in the overlapping energy range 16.20-17.98 MeV. The correction factor was equal to Fc = 0.38230.

Microscopic experimental data for the $^{27}Al(n,2n)^{26g}Al$ reaction excitation function are given in the works [5.8-5.16] and cover the neutron energy range from 13.67 MeV to 33.6 MeV. Experimental data of works [5.9-5.10], [5.12-5.16] were, in the process of analysis, corrected to the new standards for the relevant monitor reactions (see Table 2.1) and to the recommended decay data for ^{26g}Al [5.1].

The traditional activation method is very difficult to use for measurements of reaction cross section with long-lived residual nuclei due to bad statistics. Especially, near the reaction threshold where the cross section is usually small. For these purposes the Accelerator Mass Spectrometry (AMS) method was developed. In the AMS method the measured value is the amount of nuclei formed in the investigated reaction.

Experimental data for the $^{27}Al(n,2n)^{26g}Al$ reaction cross section given in works [5.8-5.10], [5.12] and [5.14] were measured by the traditional activation method. The cross sections presented in works [5.11], [5.13], [5.15] and [5.16] have been measured by means of the AMS method.

The publication of A. Wallner et al. [5.16] presents experimental data obtained by means of the AMS method and carried out by a cooperation of four scientific laboratories: IRK (Vienna, Austria), Khlopin Radium Institute (St. Petersburg, RF), Atomic Energy Research Institute (JAERI, Tokai, Japan), University of Tubingen (Germany FR).

The experiment at the IRK was performed by means of a T(d,n)He4 neutron source at IRK Cockcroft-Walton accelerator. Pure Al-metal samples, 99.999% enrichment, 1-mm thickness irradiation by neutrons of 14 energies in the interval 13.67 - 14.83 MeV. Nb-foils of 0.125-

mm thickness via reaction $^{93}Nb(n,2n)^{92m}Nb$ were used for the neutron fluence determination. The HP Ge detector was used for measuring the activity of Nb-samples.

The experiment at the Khlopin Radium Institute was performed by using the T(d,n)He4 neutron source of the neutron generator NG-400 at RI. Individual Al-samples, 14-mm in diameter and 1-mm thickness and Nb-samples, 14-mm in diameter and 0.5-mm thickness, were used in the measurements for neutron fluence determination via $^{93}Nb(n,2n)^{92m}Nb$ reaction. Stacks of samples in the order Nb-Al-Nb-Al-Nb were irradiated with neutrons of 10 energies in the interval 13.68 - 14.84 MeV. A Gamma spectrometer with HP Ge detector was used to measure the activity of Nb-samples.

The experiment at the Japan Atomic Energy Research Institute was performed by using a T(d,n)He4 neutron source at JAERI FNS facility. Reaction $^{93}Nb(n,2n)^{92m}Nb$ was used for neutron fluence determination. Sample arrangements were 5*Al-Nb. The samples were irradiated with 14.80 MeV neutrons. The activity of Nb-samples was measured by means of a gamma spectrometer with HP Ge detector.

The experiment at the University of Tubingen was carried out at a single-ended 3-MV Van de Graaff accelerator by using the T(d,n)He4 as a neutron source. Al-samples were 10-mm in diameter and of 1-mm thickness. Ni-foils, each 0.125-mm thick, were used for the neutron fluence determination via $^{58}Ni(n,x)^{57}Co$ reaction. The sample arrangement consisted of a stack of foils in the succession of Ni-Al-Ni-Al-Ni and was irradiated by neutrons with energies of 17 and 19 MeV. Two series of irradiation had been performed. A Gamma spectrometer with NaI detector was used for the activity measurement of Ni-samples.
.
The amount of Al-26 nuclei formed in all irradiation's was determined through accelerator mass spectrometry at the Vienna environmental research accelerator (VERA) at IRK. As a result of this collaborative experimental work the $^{27}Al(n,2n)^{26g}Al$ reaction cross section was measured in 27 energy points between 13.67 – 19.0 MeV.

Special correction was applied to the experimental data of [5.9], [5.11] and [5.12]. Experimental data of Iwasaki et al. [5.9], after correction to the new standards, were renormalized to the integral of A. Wallner et al. data in the overlapping interval 14.08 – 14.68 MeV, Fc= 0.61767. Obtained by means of the AMS method, the experimental data of Nakamura et al. [5.11] were renormalized to the integral of A. Wallner et al. data in the overlapping interval 15.30 – 19.00 MeV, Fc= 1.27217. Corrected to the new standards, experimental data of Ikeda et al. [5.12] were renormalized to a value of (29.50 ± 2.00) mb at 14.80 MeV, obtained by means of the AMS method in the JAERI activation [5.16].

Excitation functions for the $^{27}Al(n,2n)^{26m}Al$ and $^{27}Al(n,2n)^{26g}Al$ reactions in the energy region from threshold to 60 MeV were evaluated by means of statistical analysis of cross sections from prepared data bases.

The data base for the $^{27}Al(n,2n)^{26m}Al$ reaction was included in the analyzed experimental data [5.2-5.7] and renormalized data from EAF-2010, TENDL-2014 libraries and GNASH calculation. Cross sections from the remaining three sources were renormalized to the experimental data at 20 MeV.

The data base for the $^{27}Al(n,2n)^{26g}Al$ reaction was formed from the analyzed experimental data [5.8-5.9], [5.11-5.12] and [5.15-5.16]. In addition to the experimental information in the

data base for evaluation, renormalized data from EAF-2010, TENDL-2014 libraries and GNASH calculation were included. The EAF-2010, TENDL-2014 and GNASH data were renormalized at 20 MeV to the experimental data and covered a wide energy range of 20 – 60 MeV. Experimental data from works [10], [13] were rejected due to inconsistency with the main bulk of measured cross sections. Neither were data given in Ref. [14] taken into account in the evaluation. The authors of [14] consider that the final results of their investigation are presented in the Ref. [16]. It should be noted that cross sections measured by Filatenkov and Chuvaev by means of the traditional activation method [14] and determined through the AMS method [16] agree well in the interval between 14.1 – 14.8 MeV. Near threshold (13.6-13.74 MeV), cross-sections measured by means of the traditional activation method are significantly overestimated.

The evaluation of the $^{27}Al(n,2n)^{26m}Al$ and $^{27}Al(n,2n)^{26g}Al$ reaction excitation functions by statistical analysis of cross sections from described data bases was carried out by means of the PADE-2 code [17]. A rational function was used as the model function [18].

Uncertainties in the evaluated $^{27}Al(n,2n)^{26g}Al$ reaction excitation function are given in the form of a relative covariance matrix for the 41-neutron energy groups (LB=5). Eigenvalues of the 6-digit relative covariance matrix given in the 40-file are the following:

```
2.29451E-08    2.34306E-08    2.41623E-08    2.48190E-08
2.61715E-08    2.78413E-08    2.92478E-08    3.19992E-08
3.34173E-08    3.61163E-08    3.95786E-08    4.33685E-08
4.89694E-08    5.29808E-08    6.15034E-08    7.51562E-08
8.32315E-08    1.03520E-07    1.26658E-07    1.49217E-07
1.61588E-07    1.88452E-07    2.18906E-07    2.53236E-07
2.86156E-07    3.11811E-07    3.24717E-07    3.59672E-07
3.81217E-07    4.18547E-07    5.87033E-07    1.45516E-06
1.70287E-05    1.30565E-04    2.22458E-03    4.46645E-03
1.51400E-02    3.00939E-02    5.65327E-02    8.41867E-02
2.53924E-01
```

Uncertainties in the evaluated $^{27}Al(n,2n)^{26m}Al$ reaction excitation function are also given in the form of a relative covariance matrix for the 41-neutron energy groups (LB=5). Eigenvalues of the 6-digit relative covariance matrix given in the 40-file are the following:

```
5.36947E-06    5.56082E-06    5.83841E-06    6.26166E-06
6.82485E-06    7.34476E-06    7.92795E-06    8.75227E-06
9.68138E-06    1.04867E-05    1.21433E-05    1.40555E-05
1.54796E-05    1.79292E-05    2.09437E-05    2.34731E-05
2.54663E-05    2.89584E-05    3.27326E-05    3.73544E-05
4.17544E-05    4.53436E-05    4.84627E-05    5.28654E-05
5.64585E-05    6.52752E-05    1.01849E-04    2.87553E-04
1.11554E-03    6.53838E-03    2.86345E-02    7.60055E-02
7.77815E-02    1.10247E-01    1.41963E-01    2.21266E-01
3.47854E-01    8.69699E-01    4.09302E+00    1.89582E+01
1.15451E+02
```

Covariance matrixes of 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-40 were calculated by means of the PADE-2 code and tested independently with the COVEIG code.

Group cross sections and their uncertainties for the re-evaluated $^{27}Al(n,2n)^{26g}Al$ reaction excitation function are listed in Table 5.1. Group boundaries are the same as in File-40.

TABLE 5.1. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{27}Al(n,2n)^{26g}Al$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 60 MeV.

Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)	Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)
13.546 - 14.000	2.839E+0	4.90	25.000 - 26.000	9.703E+1	8.86
14.000 - 14.200	7.960E+0	3.44	26.000 - 27.000	9.128E+1	9.24
14.200 - 14.400	1.354E+1	2.81	27.000 - 28.000	8.563E+1	9.45
14.400 - 14.600	1.971E+1	2.47	28.000 - 29.000	8.024E+1	9.58
14.600 - 14.800	2.611E+1	2.41	29.000 - 30.000	7.520E+1	9.71
14.800 - 15.000	3.251E+1	2.61	30.000 - 32.000	6.844E+1	10.01
15.000 - 15.500	4.318E+1	3.50	32.000 - 34.000	6.071E+1	10.73
15.500 - 16.000	5.692E+1	5.00	34.000 - 36.000	5.436E+1	11.63
16.000 - 16.500	6.866E+1	6.03	36.000 - 38.000	4.915E+1	12.48
16.500 - 17.000	7.865E+1	6.49	38.000 - 40.000	4.486E+1	13.14
17.000 - 17.500	8.709E+1	6.56	40.000 - 42.000	4.131E+1	13.58
17.500 - 18.000	9.418E+1	6.44	42.000 - 44.000	3.833E+1	13.84
18.000 - 18.500	1.000E+2	6.29	44.000 - 46.000	3.582E+1	14.04
18.500 - 19.000	1.048E+2	6.17	46.000 - 48.000	3.368E+1	14.30
19.000 - 19.500	1.085E+2	6.08	48.000 - 50.000	3.185E+1	14.74
19.500 - 20.000	1.111E+2	6.02	50.000 - 52.000	3.027E+1	15.47
20.000 - 21.000	1.133E+2	5.99	52.000 - 54.000	2.889E+1	16.54
21.000 - 22.000	1.137E+2	6.21	54.000 - 56.000	2.768E+1	17.96
22.000 - 23.000	1.115E+2	6.76	56.000 - 58.000	2.661E+1	19.70
23.000 - 24.000	1.076E+2	7.53	58.000 - 60.000	2.566E+1	21.71
24.000 - 25.000	1.026E+2	8.28			

Uncertainties in the evaluated the $^{27}Al(n,2n)^{26g}Al$ excitation function range from 2.41% to 21.71%. The smallest uncertainties in the evaluated cross sections 2.41-2.81% are observed in the neutron energy range 14.2 - 15.0 MeV. Uncertainties in the $^{27}Al(n,2n)^{26g}Al$ reaction cross section exceeding 10% are observed in the energy range 30 - 60 MeV.

The evaluated excitation function for the $^{27}Al(n,2n)^{26g}Al$ reaction in the neutron energy range from threshold to 60 MeV is shown in Fig. 5.1 in comparison with the equivalent data from EAF-2010, TENDL-2014 libraries and corrected experimental data. The same information but in the narrow neutron energy range from threshold to 20 MeV is shown in Fig. 5.2. It is evident from Figs 5.1 and 5.2. that the excitation function obtained from the new evaluation agrees better with the corrected experimental data than compared with equivalent data from the EAF-2010 and TENDL-2014 libraries.

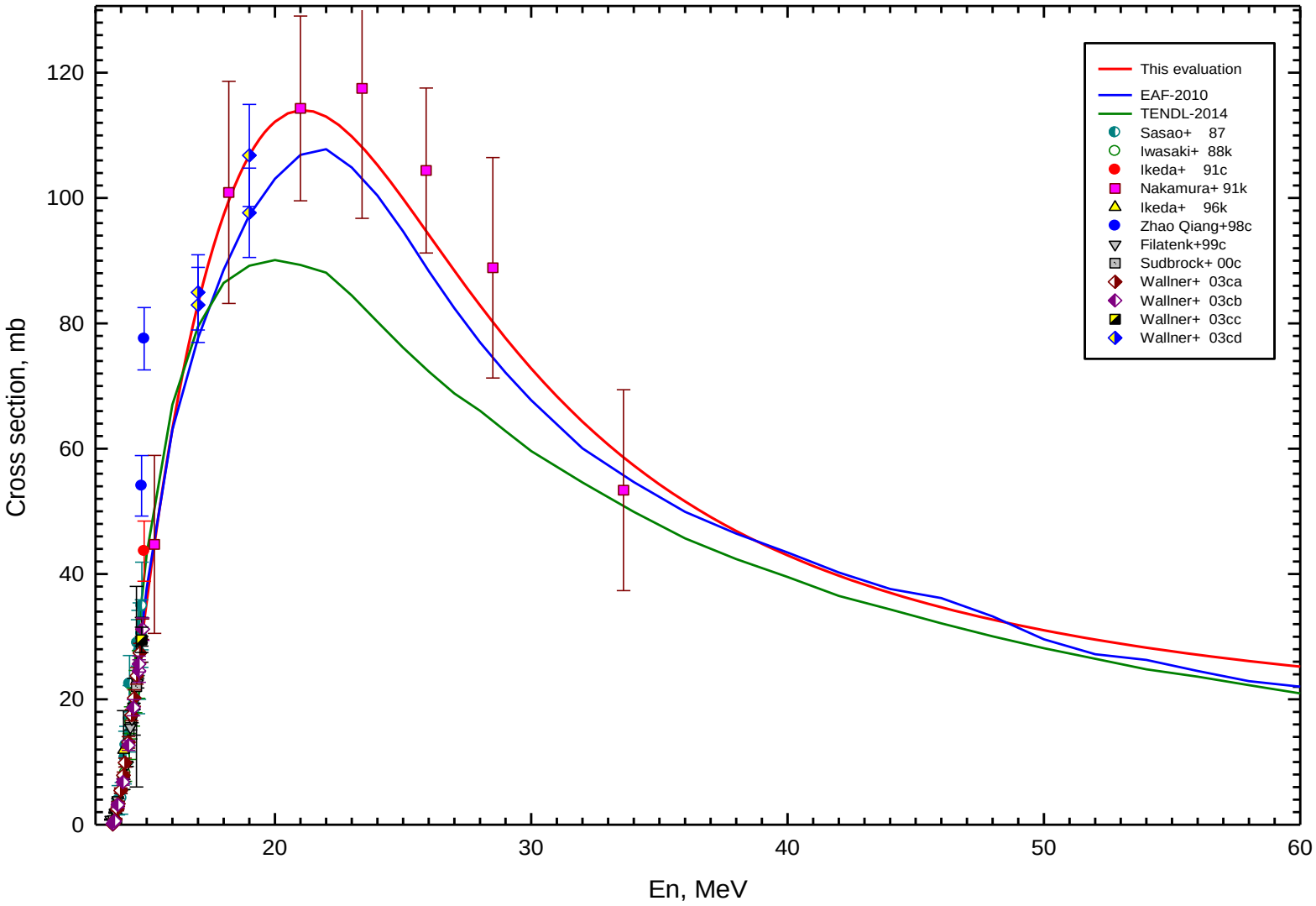

FIG. 5.1. Evaluated $^{27}Al(n,2n)^{26g}Al$ reaction excitation function in the energy range from threshold to 60 MeV in comparison with equivalent data from EAF-2010, TENDL-2014 libraries and corrected experimental data.

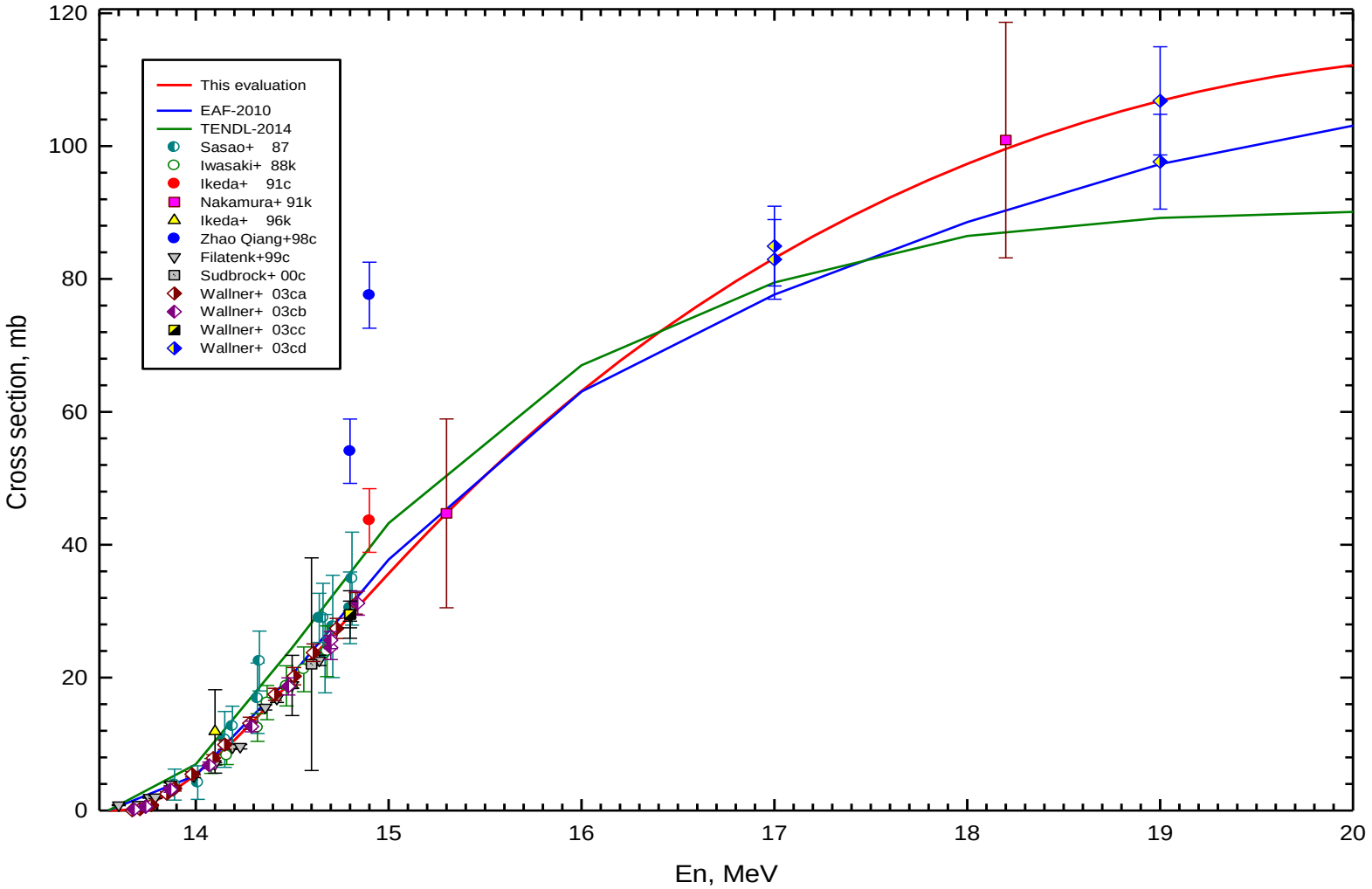

FIG. 5.2. Evaluated $^{27}Al(n,2n)^{26g}Al$ reaction excitation function in the energy range from threshold to 20 MeV in comparison with equivalent data from EAF-2010, TENDL-2014 libraries and corrected experimental data.

Group cross sections and their uncertainties for the re-evaluated $^{27}Al(n,2n)^{26m}Al$ reaction excitation function are listed in Table 5.2. Group boundaries are the same as in File-40.

TABLE 5.2. EVALUATED CROSS SECTIONS AND THEIR UNCERTAINTIES FOR THE $^{27}Al(n,2n)^{26m}Al$ REACTION IN THE NEUTRON ENERGY RANGE FROM THRESHOLD TO 60 MeV.

Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)	Neutron energy (MeV) from to	Cross section (mb)	Uncer-tainty (%)
13.775 - 14.000	1.331E-2	56.45	25.000 - 26.000	5.994E+1	12.08
14.000 - 14.200	3.466E-2	28.96	26.000 - 27.000	5.714E+1	12.19
14.200 - 14.400	7.495E-2	24.95	27.000 - 28.000	5.431E+1	12.25
14.400 - 14.600	1.619E-1	16.70	28.000 - 29.000	5.157E+1	12.23
14.600 - 14.800	3.201E-1	12.33	29.000 - 30.000	4.897E+1	12.15
14.800 - 15.000	5.730E-1	11.73	30.000 - 32.000	4.543E+1	11.94
15.000 - 15.500	1.341E+0	11.65	32.000 - 34.000	4.127E+1	11.71
15.500 - 16.000	3.222E+0	11.53	34.000 - 36.000	3.774E+1	11.66
16.000 - 16.500	6.281E+0	12.26	36.000 - 38.000	3.471E+1	11.87
16.500 - 17.000	1.060E+1	13.15	38.000 - 40.000	3.212E+1	12.36
17.000 - 17.500	1.615E+1	13.50	40.000 - 42.000	2.986E+1	13.09
17.500 - 18.000	2.270E+1	13.32	42.000 - 44.000	2.790E+1	13.98
18.000 - 18.500	2.984E+1	12.97	44.000 - 46.000	2.617E+1	14.98
18.500 - 19.000	3.709E+1	12.77	46.000 - 48.000	2.464E+1	16.04
19.000 - 19.500	4.396E+1	12.80	48.000 - 50.000	2.328E+1	17.13
19.500 - 20.000	5.005E+1	12.96	50.000 - 52.000	2.206E+1	18.21
20.000 - 21.000	5.698E+1	13.02	52.000 - 54.000	2.096E+1	19.28
21.000 - 22.000	6.266E+1	12.78	54.000 - 56.000	1.996E+1	20.32
22.000 - 23.000	6.469E+1	12.32	56.000 - 58.000	1.905E+1	21.32
23.000 - 24.000	6.428E+1	12.03	58.000 - 60.000	1.822E+1	22.29
24.000 - 25.000	6.246E+1	11.98			

Uncertainties in the evaluated the $^{27}Al(n,2n)^{26m}Al$ excitation function range from 11.53% to 56.45%. The biggest uncertainty in the evaluated cross sections of 56.45% is near threshold. Uncertainties in the $^{27}Al(n,2n)^{26m}Al$ reaction cross sections exceeding 20% are observed in the energies ranges 14.0 - 14.4 MeV and 54 - 60 MeV.

The evaluated excitation function for the $^{27}Al(n,2n)^{26m}Al$ reaction in the neutron energy range from threshold to 60 MeV is shown in Fig. 5.3 in comparison with the equivalent data from the EAF-2010, TENDL-2014 libraries and corrected experimental data. The same information, but in the narrow neutron energy range from threshold to 20 MeV is shown in Fig. 5.4. It is evident from Figs 5.3 and 5.4. that the $^{27}Al(n,2n)^{26m}Al$ reaction excitation function obtained from the new evaluation agrese much better with corrected experimental data than with the compared equivalent data from the EAF-2010 and TENDL-2014 libraries.

Integral experiments for the $^{27}Al(n,2n)^{26g}Al$ reaction were only described in the work of T. Sato and T. Suzuki [5.17]. The integral cross section of $^{27}Al(n,2n)^{26g}Al$ reaction was measured in the core of the reactor JRR-2 at JAERI by means of the activation method. The Al_2-O_3 sample was irradiated in neutron field which was classified as fission spectrum. The neutron flux was determined by an activation detector using the $^{54}Fe(n,p)^{54}Mn$ reaction.

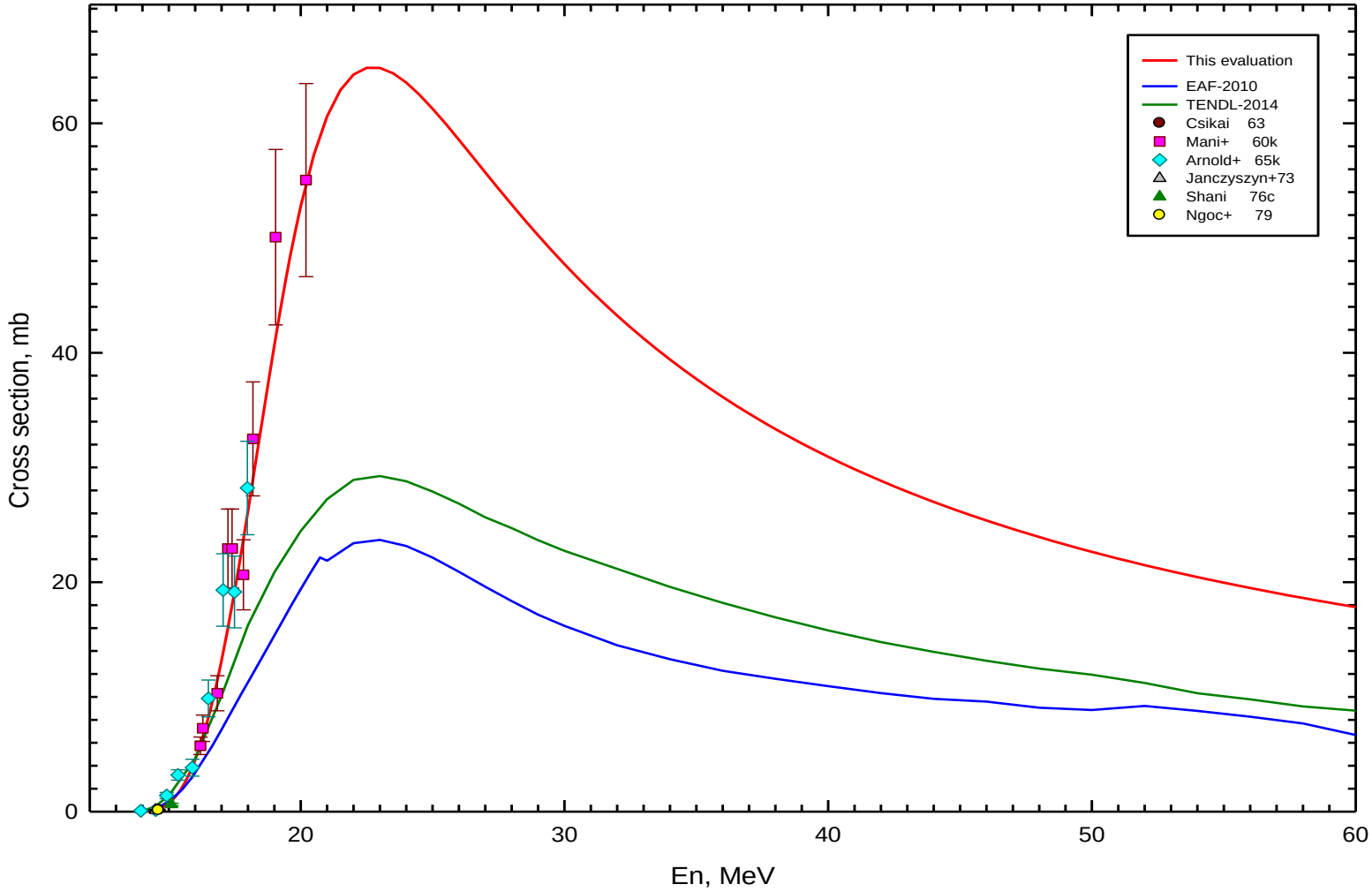

FIG. 5.3. Evaluated $^{27}Al(n,2n)^{26m}Al$ reaction excitation function in the energy range from threshold to 60 MeV in comparison with equivalent data from EAF-2010, TENDL-2014 libraries and corrected experimental data.

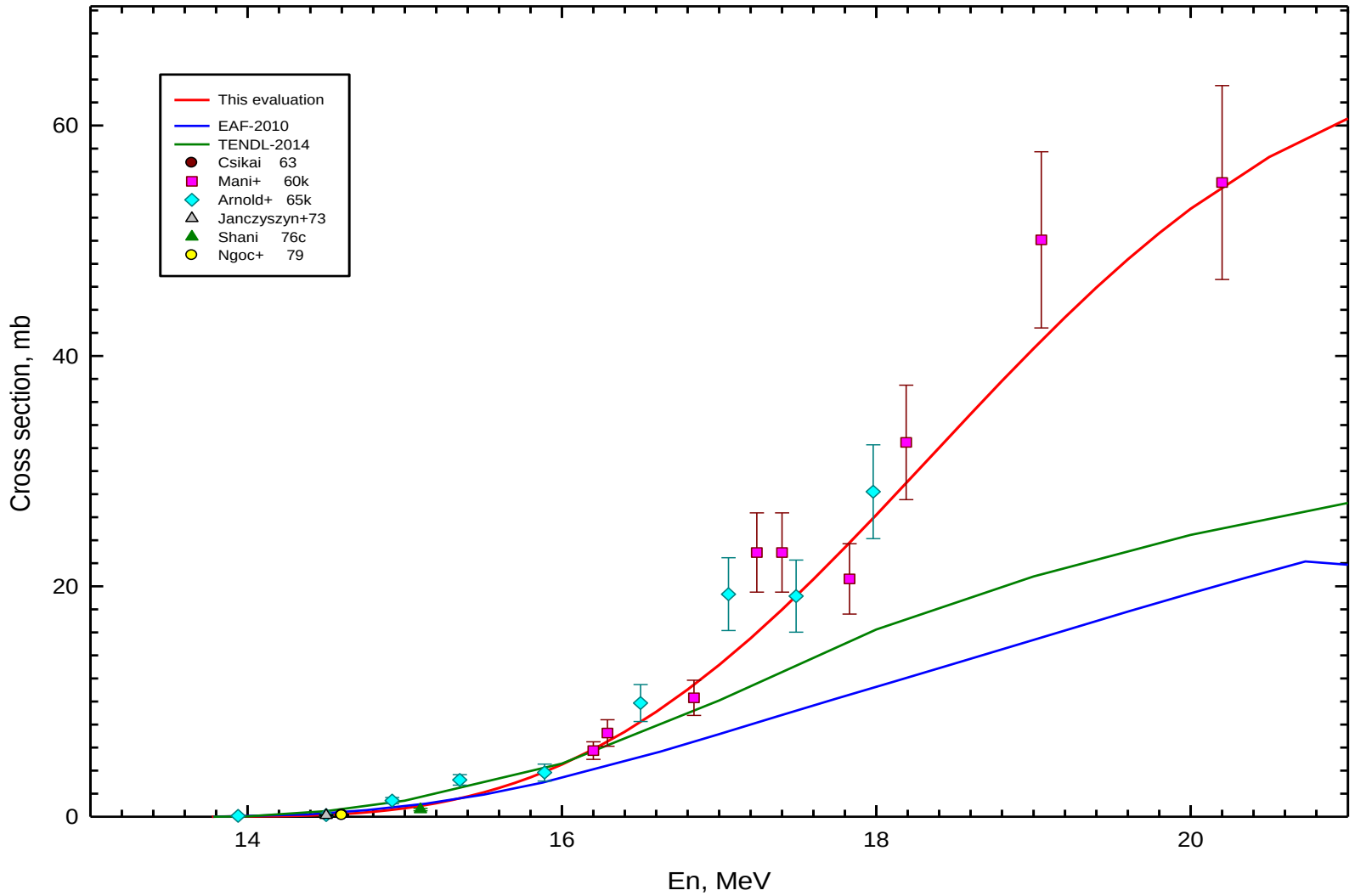

Fig. 5.4. Evaluated $^{27}Al(n,2n)^{26m}Al$ reaction excitation function in the energy range from threshold to 20 MeV in comparison with equivalent data from EAF-2010, TENDL-2014 libraries and corrected experimental data.

The original experimental value of T. Sato and T. Suzuki was corrected to the new standards for monitor reactions, Fc= 0.94084. Authors are not giving the uncertainty in the measured cross section. The uncertainty was evaluated in the process of analysis of this experiment.

The evaluated excitation function for the $^{27}Al(n,2n)^{26g}Al$ reaction was tested against the integral experimental data of T. Sato and T. Suzuki [5.17]. The calculated averaged cross sections over the ^{235}U thermal and ^{252}Cf spontaneous fission neutron spectra are compared with the EAF-2010, TENDL-2014 and experimental data in Table 5.3.

TABLE 5.3. CALCULATED AND MEASURED AVERAGED CROSS SECTIONS FOR THE $^{27}Al(n,2n)^{26g}Al$ REACTION IN ^{235}U THERMAL FISSION AND ^{252}CF SPONTANEOUS FISSION NEUTRON SPECTRA.

Type of neutron field	Averaged cross section, μb		90% response function, MeV	*C/E*
	Calculated	Measured		
^{235}U thermal fission neutron spectrum	2.4190 [A]	(3.760 ± 0.752) [5.17]	14.20 - 19.00	0.64335
	2.3864 [B]		14.20 - 18.90	0.63468
	2.5966 [C]		14.10 - 18.70	0.69059
^{252}Cf spontaneous fission neutron spectrum	6.9740 [A]		14.20 - 19.70	
	6.8310 [B]		14.20 - 19.60	
	7.3234 [C]		14.20 - 19.30	

[A] - Present evaluation
[B] - EAF-2010
[C] - TENDL-2014

The *C/E* values obtained show a very big discrepancy between calculated integral cross sections and measured cross sections for the $^{27}Al(n,2n)^{26g}Al$ reaction. The big discrepanciesmay be explained only by a one reason, namely that the neutron spectrum in the core of the reactor JRR-2 differs significantly from the ^{235}U thermal fission neutron spectrum.

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6. CONCLUSION

New evaluations of cross sections and their uncertainties have been carried out for four reactions: $^{23}Na(n,2n)^{22}Na$, $^{23}Na(n,\gamma)^{24}Na$, $^{27}Al(n,2n)^{26m}Al$ and $^{27}Al(n,2n)^{26g}Al$. The $^{23}Na(n,2n)^{22}Na$, $^{27}Al(n,2n)^{26m}Al$ and $^{27}Al(n,2n)^{26g}Al$ reaction excitation functions were evaluated in the energy range from threshold to 60 MeV. The cross section of the $^{23}Na(n,\gamma)^{24}Na$ reaction was evaluated in the wide energy region 1.000E-05 eV - 20 MeV. Uncertainty in the $^{23}Na(n,\gamma)^{24}Na$ reaction excitation function is given in 88 energy intervals.

In the described re-evaluation of the $^{23}Na(n,\gamma)^{24}Na$ reaction the excitation function cross section in the energy range 1.0000E-05 eV – 459.3 keV is given via the Reich-Moore (RM) resonance parameters. The upper boundary of the resolved resonance region is limited due to threshold of inelastic scattering.

The obtained resonance parameters permits to calculate the $^{23}Na(n,tot)$, $^{23}Na(n,el)$ and $^{23}Na(n,\gamma)^{24}Na$ excitation functions in the neutron energy range 1.000E-05 eV – 459.3 keV without any additional data (background). In the RRDFF-2012 (ENDF/B-VII.1) library for

description the $^{23}Na(n,\gamma)^{24}Na$ reaction cross section in the resolved resonance region is introduced a significant background.

Excitation function of the $^{27}Al(n,2n)^{26g}Al$ reactions was evaluated for dosimetry application at first.
Re-evaluated the $^{23}Na(n,2n)^{22}Na$, $^{23}Na(n,\gamma)^{24}Na$ excitation functions and evaluated the $^{27}Al(n,2n)^{26m}Al$ and $^{27}Al(n,2n)^{26g}Al$ reaction excitation functions may be recommender for including in a new version of IRDFF library.

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