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NEUTRON CONSTANTS AND PARAMETERS

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INVESTIGATION OF THE ^{232}Th NEUTRON CROSS-SECTIONS IN RESONANCE ENERGY RANGE

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The alternative path in the development of atomic energy is the uranium-thorium cycle. In connection with this, the measurements of the ^{232}Th neutron capture and total cross-sections and its resonance self-shielding coefficients in resonance energy range are necessary because of their low accuracy. In this work, the results of the investigations of the thorium-232 neutron cross-sections are presented. The measurements have been carried out on the gamma-ray multisection liquid detector and neutron detector as a battery of boron counters on the 120 m flight path of the pulsed fast reactor IBR-30. As the filter samples were used the metallic disks of various thickness and diameter of 45 mm. Two plates from metallic thorium with thickness of 0.2 mm and with the square of 4.5×4.5 cm^2 were used as the radiator samples. The group neutron total and capture cross-sections within the accuracy of 2-7% in the energy range of (10 eV – 10 keV) were obtained from the transmissions and the sum spectra of γ-rays from the fourth multiplicity to the seventh one. The neutron capture group cross-sections of ^{238}U were used as the standard for obtaining of thorium ones. Analogous values were calculated on the GRUCON code with the ENDF/B-6, JENDL-3 evaluated data libraries. Within the limits of experimental errors an agreement between the experiment and calculation is observed, but in some groups the experimental values are larger than the calculated ones.

Introduction

The creation of the alternative direction in the atomic power engineering on the base of uranium-thorium cycle intends preliminary the production of the precise nuclear data for thorium-232, uranium-233 and protactinium isotopes [1,2]. Up until now, it was essential to study the neutron capture cross-section, a resonance self- shielding and their temperature dependence, e.g. Doppler effect in neutron cross-sections, important for the nuclear safety problems solving. Now there are the experimental data, which were obtained with accuracy of 1-6%, but they have a discrepancy of up to 50%. Recently, we carried out the preliminary measurements of the radiative capture in the energy range 21.5- 215eV [3] and also transmissions and Doppler coefficients in them for thorium-232 [4].

After the modernization of the multisectional (n,γ)-detector of the Romashka type [5] and the construction of the installations on the base of germanium detector, we started the thorium - 232 neutron capture and total cross-section measurements within 20 eV - 10 keV resonance energy region. After this, we intend to carry out similar investigations with the uranium - 233 and protactinium isotopes.

Experimental technique

The measurements of the neutron capture and total cross-sections were being carried out on the 121 m (124 m) neutron flight paths of the IBR-30 (W = 10 kW, f = 100 Hz, τ = 4 μsec) with the 16-section liquid detector of volume 80 l and the neutron boron detector. The general scheme is given in Fig. 1.

The hole of the lead collimator in front of the (n,γ)-detector was 70 mm diameter. The B_4C and Cd filters were placed in the beam for removing the background of recycled neutrons. The filter samples were made from the thorium oxide and the metallic thorium discs of various thickness. Two plates from metallic thorium with total thickness of 0.2 mm and with the square of S = 4.5×4.5 cm^2 were used as the radiator samples in the liquid detector. The U_3O_8 powder with uranium-238 (99.999%) and by weight of 3.86 g, contained in the aluminum tank with the 46 mm inside diameter, was served as the standard radiator as well. The 16-section liquid detector is described in [5]. From the moment the measurements started, the vary detector had a new lead shielding with thickness of 10 cm. The shape of the lead shielding was changed. The neutron detector in the form of a battery of three boron counters SNM-13 placed into a polyethylene disc 100 mm in diameter and 15 mm in thickness, was within 124 m from the IBR-30 fuel core moderator. The liquid detector operated in the multiplicity coincidence spectrometry conditions. Figs. 2 and 3 show typical apparatus time-of-flight spectra. The background components from natural radioactivity are low, and the effect - background ratio for the liquid detector in third multiplicity spectrum is 100% at best. The background for the boron neutron detector is 5-10%. The SNM-12 counter was used as monitor.

The processing technique and the results of measurements

Some runs of measurements were performed with the help of the liquid (n,γ)-detector having used in turn the ^{232}Th radiator sample and ^{238}U one. The measurements with the neutron boron detector were carried out, when the thorium and uranium radiator samples were not in the liquid detector. After the background components had been subtracted and the time-of-flight spectra had been brought mutual monitor coefficient, one determined a number of counts in the energy groups of the BNAB constants system [6] over the energy region of 20 eV – 10 keV. The number of counts within the energy groups are connected with a neutron flux and the neutron capture cross-section for ^{238}U and ^{232}Th of the following relations:

$$N_c^U / M^U = S^U n^U \langle\sigma_c\rangle^U \varepsilon^U(E)\varphi(E),\ N_c^{Th} / M^{Th} = S^{Th} n^{Th} \langle\sigma_c\rangle^{Th} \varepsilon^{Th}(E)\varphi(E), \quad (1)$$

where: M^U and M^{Th} are the monitor coefficients for the U and Th, S^U and S^{Th} are the areas , n^U and n^{Th} are the thickness of radiator samples , $\varepsilon^U(E)$ and $\varepsilon^{Th}(E)$ are the gamma-rays efficiencies of registration, $\varphi(E)$ is a neutron flux.

Supposing that the gamma-rays efficiency of registration for the uranium and thorium samples is the same, one may get from relations (1) the expression to determine a capture cross -section according to a know ^{238}U capture cross section:

$$\langle\sigma_c\rangle^{Th} = \langle\sigma_c\rangle^U M^U S^U n^U N_c^{Th} / M^{Th} S^{Th} n^{Th} N_c^U \quad (2)$$

The mentioned liquid detector made it possible to define the time-of-flight spectra for 16 multiplicity of coincidences. However, as there had been observed the influence of scattered neutrons in the first three multiplicity, the summation of the spectra from the fourth multiplicity to the seventh one was performed. At this stage, a neutron capture was not observed with the eighth multiplicity spectrum.

After that the grouping of total spectra was being done, and the ^{232}Th - group capture cross-section, according to formula (2), was being determined. The group capture cross-sections of the ^{238}U were obtained by the calculation according to the GRUKON program [7] on the basis of the evaluated data of various libraries.

The experimental and calculated values of the group neutron capture cross-sections for thorium-232 and uranium-238 are shown in Table 1. The experimental errors of the capture cross sections are about 7-10%. It is seen from Table 1 that within the limits of experimental errors there is observed an agreement between the experiment and calculation, but in the group 17 an experimental value is larger than the calculated one.

The total transmissions were measured on three ^{232}Th filter samples with thickness of 0.01303 at/b, 0.02655 at/b and 0.0517 at/b. The group transmissions and observed cross-sections were determined by following formulas:

$$T_t(n, E, \theta) = \int_{\Delta E} \varphi(E)\varepsilon(E)e^{-\sigma(E,\theta)n}dE / \int_{\Delta E} \varphi(E)\varepsilon(E)dE = (N_s - F_s)M/(N_{o/b} - F_{o/b}),\ \sigma_t^{ob} = (-\ln T_t)/n, \quad (3)$$

where: $\varphi(E)$ is the neutron flux, $\varepsilon(E)$ is the detector efficiency, σ_t is the total cross-section, n is the thickness of the filter sample, E is the neutron energy , θ is the temperature of the filter sample, N_s and $N_{o/b}$ are the detector counts with sample and without it, F_s and $F_{o/b}$ are the detector backgrounds, M is the monitor coefficient, σ_t^{ob} is the observed total cross-section.

The group total cross-sections were obtained by extrapolating the observed experimental ones to zero thickness of the filter sample, i. e. by multiplication of the observed cross-sections by the calculated coefficients of the self-shielding:

$$\langle\sigma_t\rangle^{exp} = K_{sh}\sigma_t^{ob}(n),\ K_{sh} = \langle\sigma_t\rangle^{cal}/\sigma_t^{cal}(n), \quad (4)$$

where: $\langle\sigma_t\rangle^{exp}$ and $\langle\sigma_t\rangle^{cal}$ are the experimental and calculated group total cross-sections, $\sigma_t^{ob}(n)$ and $\sigma_t^{cal}(n)$ are the experimental and calculated the group observed total cross-sections for the filter sample n at/b thick.

The experimental and calculated group total cross-section for thorium-232 are shown in Table 2.

The group transmissions and cross-sections are averaged over Fermi spectrum. The experimental uncertainties of transmissions are 0.2–0.5%, the errors of the total cross-sections are 2-10%. The total transmissions are usually measured at $n\sigma_t$ = 0.2–0.4. This leads to underestimating the average group cross-sections by 20-40% in the region of unresolved resonance, if the correction for the resonance self-shielding of the averaged cross-sections is not introduced.

Conclusion

The measurements of the neutron capture cross-sections and transmissions were carried out and, using them the group total cross-sections of the ^{232}Th in the 10 eV – 10 keV energy region were determined. The analogous integral characteristics for the ^{232}Th were obtained on the basis of the evaluated data of ENDF/B-6 and JENDL-3.

In future, we intend to continue work in this direction to determine self-indication functions in neutron capture and on their base to define the resonance self-shielding factors.

Table 1.

The experimental and calculated neutron capture cross-sections of the ^{232}Th and ^{238}U

N_{gr}	E_{gr}, keV	$<\sigma_c>^{Th}$. b	$<\sigma_c>^{Th}$ [8]	$<\sigma_c>^{Th}$ [9]	$<\sigma_c>^{U}$ [8]
12	10.0 - 4.65	0.86 ± 0.10	0.867	0.899	0.814
13	4.65 - 2.15	2.04 ± 0.20	1.421	1.19	1.24
14	2.15 - 1.0	2.34 ± 0.20	2.104	1.79	1.70
15	1.0 - 0.465	3.96 ± 0.40	3.380	2.75	3.32
16	465 - 215 eV	5.79 ± 0.60	8.62	7.60	4.55
17	215 - 100	20.67 ± 3.00	14.53	14.68	20.3
18	100 - 46.5	19.24 ± 2.00	21.91	21.80	16.6
19	46.5 - 21.5	57.00 ± 5.00	54.0	54.4	54.2
20	21.5 - 10	-	0.662	0.641	0.841

Table 2.

The experimental and calculated total transmissions and cross-sections of the ^{232}Th

N_{gr}	E_{gr} (keV)	T_t (5 mm)	T_t (10 mm)	T_t (20 mm)	$<\sigma_t>$ (b)	$<\sigma_t>$ [8]	$<\sigma_t>$ [9]
12	10.0 - 4.65	0.804	0.648	0.443	17.7 ± 0.8	16.4	17.3
13	4.65 - 2.15	0.792	0.651	0.444	19.6 ± 0.9	19.0	19.9
14	2.15 - 1.0	0.794	0.648	0.465	21.9 ± 0.9	20.7	21.4
15	1.0 - 0.465	0.788	0.653	0.465	24.9 ± 1.1	23.9	22.1
16	465 - 215eV	0.790	0.661	0.481	33.5 ± 1.3	32.9	32.2
17	215 - 100	0.791	0.659	0.485	45.3 ± 1.5	41.8	43.6
18	100 - 46.5	0.795	0.670	0.487	71.8 ± 3.0	65.5	65.3
19	46.5 - 21.5	0.775	0.640	0.489	74.6 ± 4.0	71.9	73.2
20	21.5 - 10	0.843	0.729	0.532	12.4 ± 0.6	11.1	11.5

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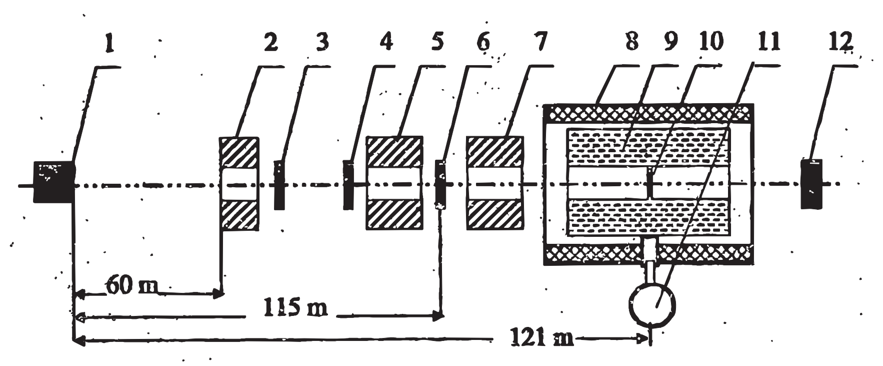

Fig. 1. The general scheme of the experiment:
1- IBR-30 fuel core moderator, 2, 5, 7 - collimators, 3 - Cd filter, 4 - monitor, 6 - Th sample - filter, 8 - lead shielding of the (n,γ)-detector, 9 - liquid detector, 10 - Th sample - radiator, 11 - HpGe-detector, 12 - neutron detector

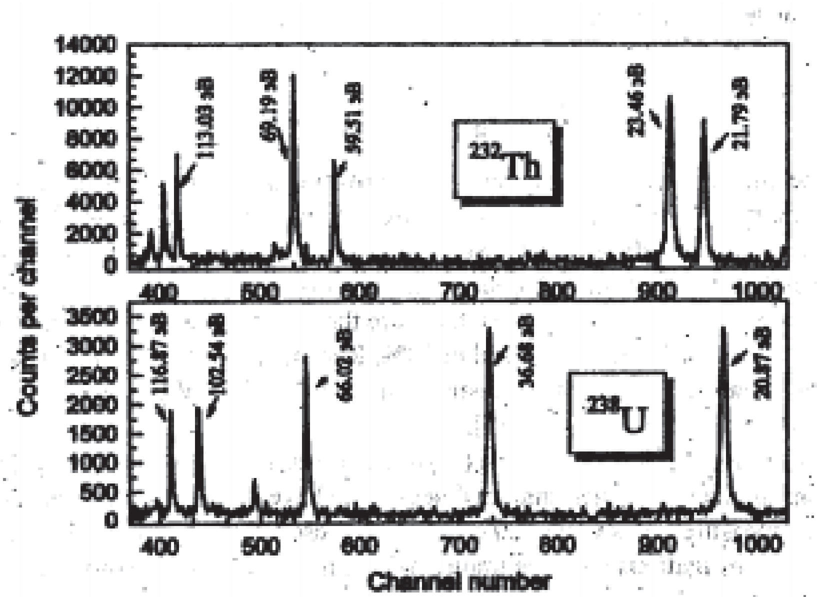

Fig. 2. The time-of-flight spectra of γ-ray coincidence summed from the 4th to 7th multiplicity for ^{232}Th and ^{238}U. (The time channel width – 2 μsec).

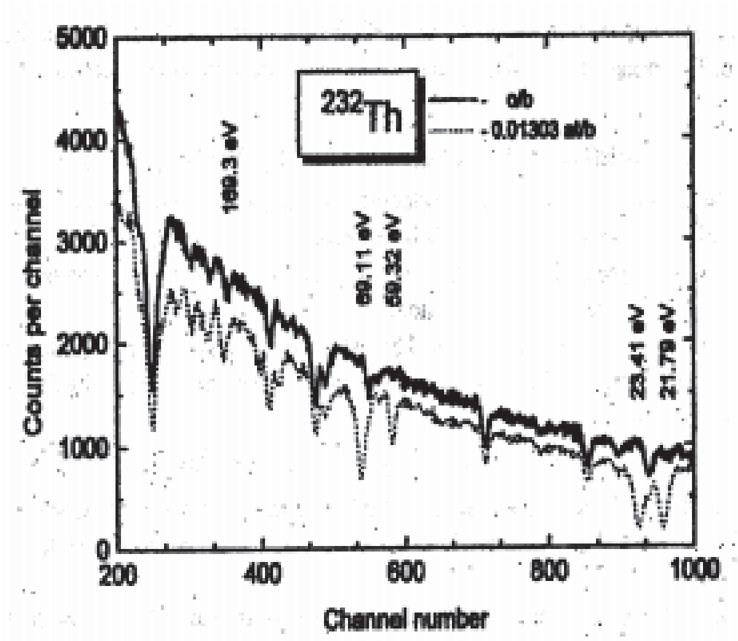

Fig. 3. The time-of-flight spectra of neutron detector for the open beam and with ^{232}Th sample -filter. (The time channel width – 2 μsec).

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ACT-1000. GROUP ACTIVATION CROSS-SECTION LIBRARY FOR WWER-1000 TYPE REACTORS

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ACT-1000 – GROUP ACTIVATION CROSS-SECTION LIBRARY FOR WWER-1000 TYPE REACTORS. The ACT-1000, a problem-oriented library of group-averaged activation cross-sections for WWER-1000 type reactors, is based on evaluated microscopic cross-section data files. The ACT-1000 data library was designed for calculating induced activity for the main dose-generated nuclides contained in WWER-1000 structural materials. In preparing the ACT-1000 library, 47 group-averaged cross-section data for the 10^{-9}-17.33 MeV energy range were used to calculate the spatial-energy neutron flux distribution.

New radiation safety rules substantially reduce the dose limits for occupational exposure during reactor facility operation. Modern problem-oriented activation reaction cross-sections are needed to substantiate the permissible radiation levels. In addition, these activation cross-sections can be used to make very accurate assessments of the amounts of radioactive waste generated, as well as their activity, in the decommissioning planning stage.

The ACT-1000 library contains activation cross-sections of the main nuclides in the WWER-1000 reactor facility construction materials determining the radiation situation during operation (power operation, inspection of shut-down equipment, transport-process operations) in a 47-group representation in the 10^{-9}-17.33 MeV neutron energy range.

1. Compilation of the activation cross-section library

The cross-section library for activation calculation contains evaluated data for 17 activation reactions. It includes seven (n,p), four capture and four (n,2n) reactions, and one inelastic scattering and one (n,α) reaction.

When the library was set up, evaluated data were examined in all the national evaluated data libraries, both general (ENDF/B-VI, BROND-2, JENDL-3.2, JEF-3.1, CENDL-2, FENDL-2) and specialized (International Reactor Dosimetry File (IRDF-90v2), Russian National Library of Activation Cross-Sections (ADL-3), European Library of Activation Cross-Sections (EAF-99), two special-purpose Japanese libraries - the JENDL/A-3.2 data file for activation calculation, and the new JENDL/D-99 dosimetry file), and compared with all the experimental data, including the latest precision measurements. The evaluations for inclusion in the library of recommended cross-sections were selected on the basis of a critical analysis of the data examined. Table 1 gives a list of the reactions examined and the sources of the evaluations selected together with their authors. For each reaction, the recommended activation cross-sections compared with the experimental data are shown in Figs 1 to 17.

Table 1

List of reactions

Isotope/Reaction	MAT	Threshold energy (MeV)	Library	Authors	Date
$^{16}O(n,p)^{16}N$	825	10.245	ENDF/B-VI	G. Hale, Z. Chen, P. Young	January 1990
$^{17}O(n,p)^{17}N$	828	8.3658	BROND-3	K.I. Zolotarev, A.B. Pashchenko,	July 2000
$^{46}Ti(n,p)^{46}Sc$	2212	1.6197	RRDF-98	K.I. Zolotarev, S. Badikov, A.B. Pashchenko	January 1994
$^{48}Ti(n,p)^{48}Sc$	2232	3.2756	RRDF-98	K.I. Zolotarev, A.B. Pashchenko	December 1993
$^{50}Cr(n,\gamma)^{51}Cr$	2425		ENDF/B-6	D.C. Hetrick	October 1997
$^{52}Cr(n,2n)^{51}Cr$	2431	12.273	IRDF-90	H. Vonach	1990
$^{54}Fe(n,p)^{54}Mn$		0.087	RRDF-98	K.I. Zolotarev, V.N. Manokhin	August 1996
$^{58}Fe(n,\gamma)^{59}Fe$	2637		ENDF/B-6	N. Larson et al.	Rev.2 – September 1998
$^{59}Co(n,\gamma)^{60m+g}Co$	2725		ENDF/B-6	A. Smith et al.	Rev.1 – June 1992
$^{59}Co(n,2n)^{58}Co$	2725	10.632	JENDL/D-99	S. Iwasaki, T. Watanabe	August 1996
$^{58}Ni(n,p)^{58}Co$	2825	0.4078	RRDF-98	K.I. Zolotarev	February 1997
$^{58}Ni(n,2n)^{57}Ni$	2825	12.431	JENDL/D-99	K. Kobayashi	November 1997
$^{60}Ni(n,p)^{60}Co$	2821	2.0758	RRDF-98	K.I. Zolotarev	November 1996
$^{63}Cu(n,\alpha)^{60}Co$	2911	10^{-11}	RRDF-98	K.I. Zolotarev	March 1997
$^{93}Nb(n,n')^{93m}Nb$	4112	0.0307	RRDF-98	K.I. Zolotarev, S. Badikov	August 1996
$^{93}Nb(n,2n)^{92m}Nb$	4112	9.0523	RRDF-98	K.I. Zolotarev, S. Badikov	August 1996
$^{98}Mo(n,\gamma)^{99}Mo$	4243		ENDF/B-6	Schenter, Schmittroth et al.	February 1980

As the Table and the Figures show, the evaluated cross-sections for 8 reactions are most reliably presented in the new version of the national dosimetry file, RRDF-98 [1]. The evaluated data for two reactions are taken from the updated Japanese dosimetric file, JENDL/D-99 [2], five evaluations from the ENDF/B-6 and one recommended curve each from IRDF-90v2 [3] and BROND-3. The data for the $^{58}Fe(n,\gamma)^{59}Fe$ and $^{59}Co(n,\gamma)^{60m+g}Co$ reaction cross sections from the ENDF/B-6 library are included in the International Reactor Dosimetry File, IRDF-90v2.

All the selected cross-sections were averaged over the ^{252}Cf spontaneous fission neutron spectrum [4] and the ^{235}U thermal fission neutron spectrum [5]. In order to verify the reliability of the selected evaluations, the average cross-sections from the compiled library are compared in Table 2 with integral experimental data (if such data exist), and also with the corresponding average cross-sections from IRDF-90v2.

Table 2

Measured and calculated cross-sections averaged over the ^{252}Cf spontaneous fission neutron spectrum and the ^{235}U thermal fission neutron spectrum

	^{252}Cf			^{235}U		
Reaction	RRDF-98	IRDF-90	Experiment	RRDF-98	IRDF-90	Experiment
	$\langle\sigma\rangle$, mb	$\langle\sigma\rangle$, mb	$\langle\sigma\rangle$, mb	$\langle\sigma\rangle$, mb	$\langle\sigma\rangle$, mb	$\langle\sigma\rangle$, mb
^{48}Ti(n,p)^{48}Sc	0.3979	0.3864	0.428 ± 0.008 [6]	0.2833	0.2749	0.302 ± 0.010 [6]
^{63}Cu(n,α)^{60}Co	0.6803	0.6778	0.6897 ± 0.0130 [6]	0.5228	0.5214	0.5271 ± 0.0139*
^{93}Nb(n,2n)^{93m}Nb	0.7701	0.7773	0.749 ± 0.038 [6]	0.4416	0.4459	0.4576 ± 0.0226*
^{93}Nb(n,n')^{92m}Nb	146.02	142.55	147.5 ± 2.5 *	143.46	139.97	147.6 ± 7.0*

* - evaluated by the authors

As Table 2 shows, there are integral experimental cross-sections for only 4 of the 17 reactions examined. In all 4 cases, the agreement obtained confirms the reliability of the recommended cross-sections.

2. Processing of evaluated microscopic data for reaction cross-sections

The recommended source data for the cross-sections of all 17 activation reactions are presented in the ENDF-6 format. The data for the threshold reaction cross-sections are presented in point form with the given interpolation laws in the neutron energy range from threshold to 20 MeV. The data for the neutron radiative capture reaction cross-sections in the resolved resonance region (RRR) and the unresolved resonance region (URR) are given as the corresponding resonance parameters. Starting from the right-hand boundary of the URR and up to 20 MeV, the radiative capture cross-sections are given in point form with the given interpolation laws. All the original data files were first processed using the LINEAR [7] program. This helped reduce all the original point data for the cross-sections to a single linear interpolation law. The RECENT [7] program was used to reconstruct the cross-sections of the neutron radiative capture reactions in the RRR and URR and to obtain point data for the entire 10^{-5} eV-20 MeV neutron energy range. The data for the reaction cross-sections were linearized and reproduced with 1% accuracy.

Following processing of the microscopic source data using the LINEAR and RECENT programs, the values obtained for the neutron radiative capture cross-sections correspond to the neutron gas temperature T = 0 K.

The ^{50}Cr(n,γ)^{51}Cr, ^{58}Fe(n,γ)^{59}Fe, ^{59}Co(n,γ)$^{60m+g}$Co and ^{98}Mo(n,γ)^{99}Mo reaction cross-sections were determined for four temperature values: 350 K (concrete shielding), 570 K (core shroud), 580 K (reactor vessel), 720 K (reactor core). The doppler resonance broadening was taken into account using the SIGMA1 program [7].

3. Preparation of group constants

47-group cross-sections of the 17 activation reactions producing long-lived radioactive nuclides were calculated for the six most important positions in the WWER-1000 reactor. These include the reactor core, core shroud, reactor vessel (vessel internal surface, inside the vessel, vessel external surface) and concrete shielding.

The boundaries of the 47-group break-down of the energy scale are given below in Table 3.

Table 3

Energy boundaries corresponding to the 47-group break-down

Group number	$E_{gr}(i) - E_{gr}(i+1)$, eV	Group number	$E_{gr}(i) - E_{gr}(i+1)$, eV
1	1.7330E+07 - 1.4190E+07	24	3.6880E+05 - 2.9720E+05
2	1.4190E+07 - 1.2210E+07	25	2.9720E+05 - 1.8320E+05
3	1.2210E+07 - 1.0000E+07	26	1.8320E+05 - 1.1110E+05
4	1.0000E+07 - 8.6070E+06	27	1.1110E+05 - 6.3780E+04
5	8.6070E+06 - 7.4080E+06	28	6.3780E+04 - 4.0870E+04
6	7.4080E+06 - 6.0650E+06	29	4.0870E+04 - 3.1830E+04
7	6.0650E+06 - 4.9660E+06	30	3.1830E+04 - 2.6060E+04
8	4.9660E+06 - 3.6790E+06	31	2.6060E+04 - 2.4180E+04
9	3.6790E+06 - 3.0120E+06	32	2.4180E+04 - 2.1880E+04
10	3.0120E+06 - 2.7250E+06	33	2.1880E+04 - 1.5030E+04
11	2.7250E+06 - 2.4660E+06	34	1.5030E+04 - 7.1020E+03
12	2.4660E+06 - 2.3650E+06	35	7.1020E+03 - 3.3550E+03
13	2.3650E+06 - 2.3460E+06	36	3.3550E+03 - 1.5850E+03
14	2.3460E+06 - 2.2310E+06	37	1.5850E+03 - 4.5400E+02
15	2.2310E+06 - 1.9200E+06	38	4.5400E+02 - 2.1440E+02
16	1.9200E+06 - 1.6530E+06	39	2.1440E+02 - 1.0130E+02
17	1.6530E+06 - 1.3530E+06	40	1.0130E+02 - 3.7270E+01
18	1.3530E+06 - 1.0030E+06	41	3.7270E+01 - 1.0680E+01
19	1.0030E+06 - 8.2080E+05	42	1.0680E+01 - 5.0430E+00
20	8.2080E+05 - 7.4270E+05	43	5.0430E+00 - 1.8580E+00
21	7.4270E+05 - 6.0810E+05	44	1.8580E+00 - 8.7640E-01
22	6.0810E+05 - 4.9790E+05	45	8.7640E-01 - 4.4140E-01
23	4.9790E+05 - 3.6880E+05	46	4.4140E-01 - 1.0000E-01
		47	1.0000E-01 - 1.0000E-05

The 47-group cross-sections were prepared using the GROUPIE program [7] and the evaluated microscopic data described above for the reaction cross-sections, obtained following processing with the LINEAR, RECENT and SIGMA1 programs. It should be noted that all the programs listed are recommended by the IAEA for processing nuclear data represented in the ENDF-6 format.

The cross-sections were weighted taking into account the neutron spectra in the six positions of the WWER-1000 reactor under investigation. In preparing the weighting functions, the calculated spectra data obtained using the ANISN program were taken as a basis. The spatial-energy distribution for the neutron flux density was calculated in a cylindrical geometry using the VITAMIN-E multigroup constant library with a 174-group neutron spectrum break-down in the 10^{-9}-17.33 MeV energy range. The calculations were performed in the Gidropress Special Design Office [8].

The structure of the spectra in the 10^{-5}-10^{-1} eV neutron energy range was worked out in detail on the assumption that the neutron flux density distribution, *F(E)*, was Maxwellian. Account was taken of the fact that in the core and the reactor vessel the Maxwellian distribution is substantially deformed as the neutron energy decreases owing to an increase in their absorption in the environment. In the concrete shielding the neutron spectrum in the 10^{-5}-10^{-1} eV range is sufficiently well described by Maxwellian distribution.

Owing to the large statistical errors in the calculated data for neutron spectra in the reactor vessel, concrete shielding and core shroud, values of the weighting functions greater than 12 MeV were obtained by approximation by means of the ^{235}U thermal neutron fission spectrum. The following relationship was used to calculate the ^{235}U fission spectrum values:

$$F(E) = C \times 0.7391 \times \sqrt{E} \times e^{-E/1.33}$$

where	C = 1.000	for $0 < E \leq 4.5$ MeV,
	C = 1.180-0.040×*E*	for $4.5 < E \leq 9.5$ MeV,
	C = 1.0375-0.025×*E*	for $9.5 < E \leq 20.0$ MeV.

The weighting function, corresponding to the averaged core neutron spectrum, was determined above 12 MeV on the basis of the 174-group calculation data.

The weighting functions for all six selected positions in the WWER-1000 reactor were given in the form of point values of the neutron flux density, *F(E)*, in the 10^{-5} eV-18 MeV neutron energy range.

A basic criterion in preparing the weighting functions was agreement with the calculated 174-group data to within 0.5%.

The weighting functions used in preparation of the 47-group cross-sections are shown in Figs 18 to 23 compared with the corresponding 174-group neutron spectra.

4. Format for the data in the group cross-section library

The 47-group cross-sections for all six positions in the WWER-1000 reactor obtained for the 17 activation reactions are represented in the BGL-1000 format in descending neutron energy order. Cross-section values are given for all 47 groups. For each reaction, the first cross-section value corresponds to the energy range (1.733E+07–1.419E+07) eV and the last to the range (1.000E-01–1.000E-05) eV.

The data for each reaction begin with an information line. The first 19 line positions are reserved for identification of the reaction. Positions 20 to 29 are used to indicate the name of the library from which the source microscopic data for the reaction excitation function were taken. Column 30 onwards indicates the position in the reactor for the group cross-sections given. For example, for the group data for the $^{59}Co(n,2n)^{58}Co$ reaction cross-section, obtained on the basis of the microscopic values of the reaction excitation function from the Japanese dosimetric file, the information line for the internal surface of the reactor vessel takes the form:

Co59(n,2n)Co58 JENDL/D99 KORPUS cell-(9-10-11-12)

The cross-section data for this reaction, obtained for the central part of the vessel, are accompanied by the information line:

Co59(n,2n)Co58 JENDL/D99 KORPUS cell-(13-14)

The abbreviations used for the libraries-initial sources in the information line are as given in column 4 of Table 1.

The evaluated group cross-sections in the BGL-1000 format for the 17 activation reactions, obtained for the six positions of the WWER-1000 reactor, are included in the working nuclear constant libraries of the Nuclear Data Centre, Obninsk.

The authors wish to thank Mr. V.I. Tsofin and Mr. V.V. Kal’chenko for their support and the fruitful discussions on the results obtained.

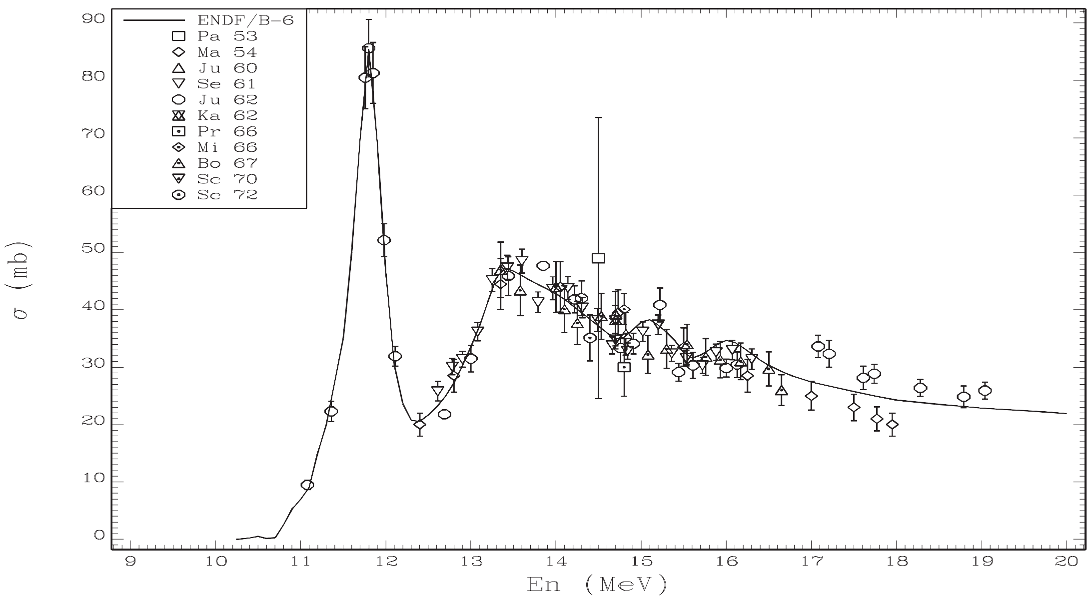

Fig. 1. Recommended $^{16}O(n,p)^{16}N$ reaction excitation function compared with experimental data

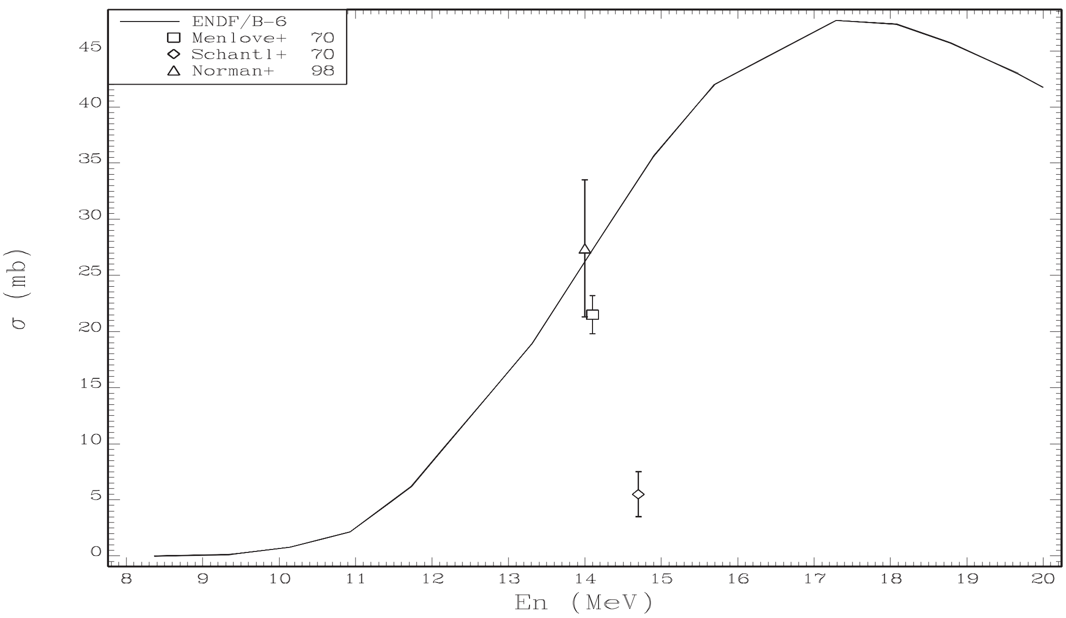

Fig. 2. Recommended $^{17}O(n,p)^{17}N$ reaction excitation function compared with experimental data

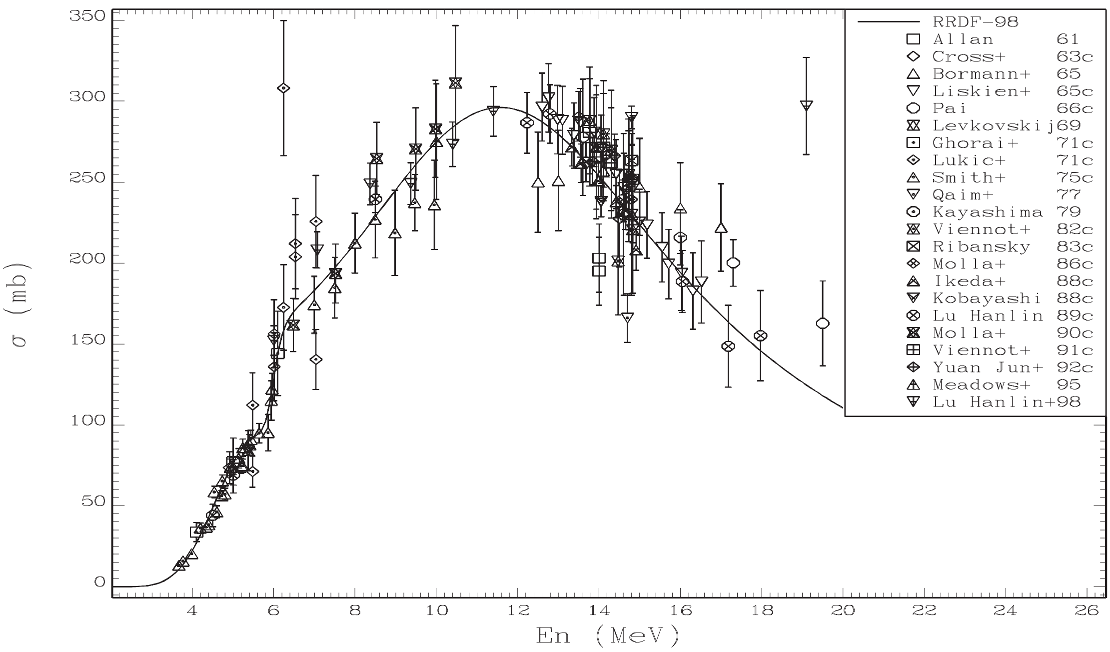

Fig. 3. Recommended ^{46}Ti(n,p)^{46}Sc reaction excitation function compared with experimental data

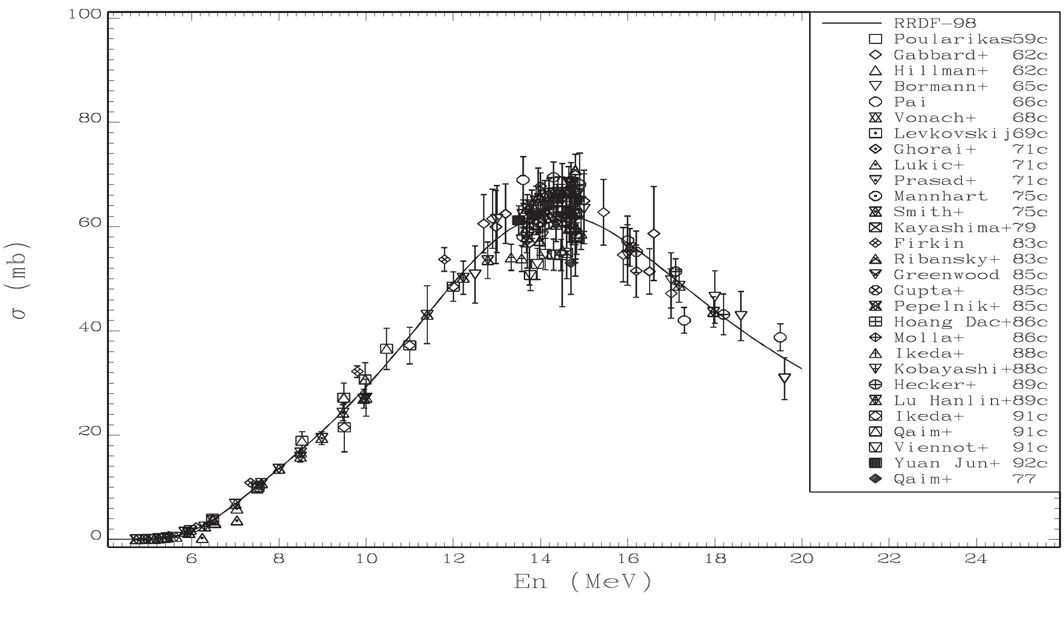

Fig. 4. Recommended ^{48}Ti(n,p)^{48}Sc reaction excitation function compared with experimental data

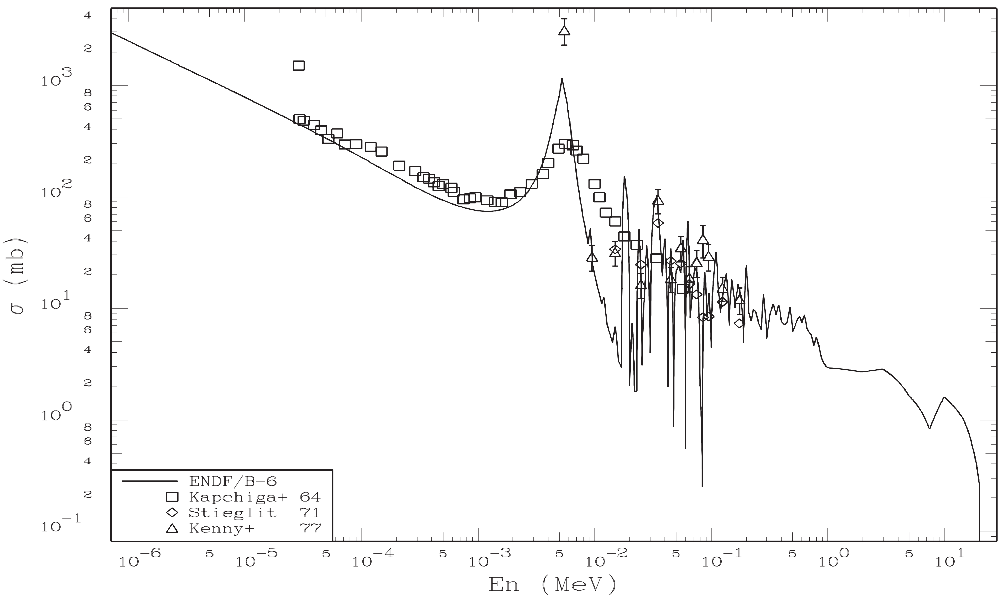

Fig. 5. Recommended $^{50}Cr(n,\gamma)^{51}Cr$ reaction excitation function compared with experimental data

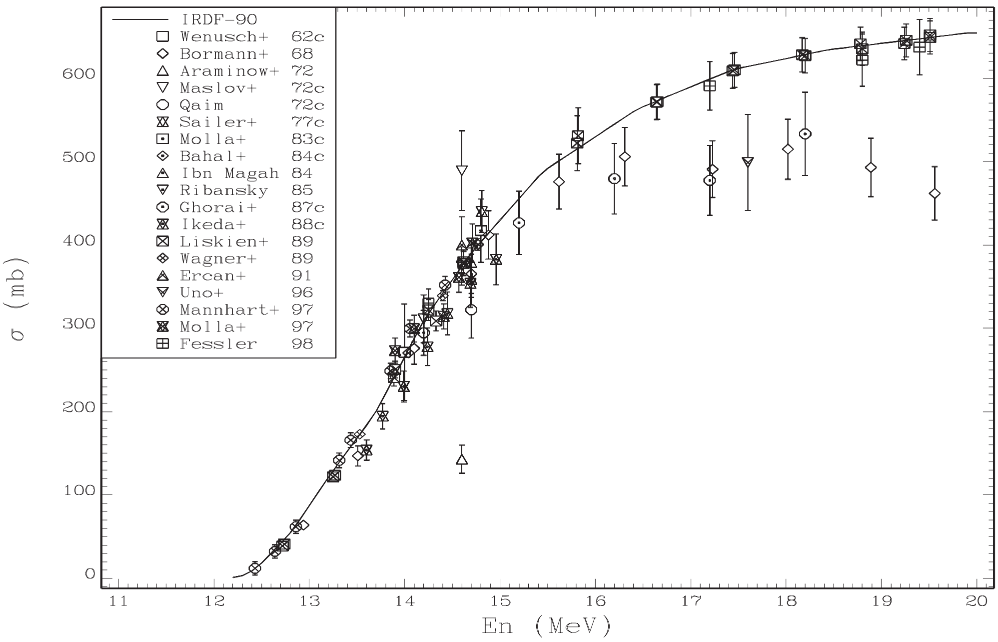

Fig. 6. Recommended $^{52}Cr(n,2n)^{51}Cr$ reaction excitation function compared with experimental data

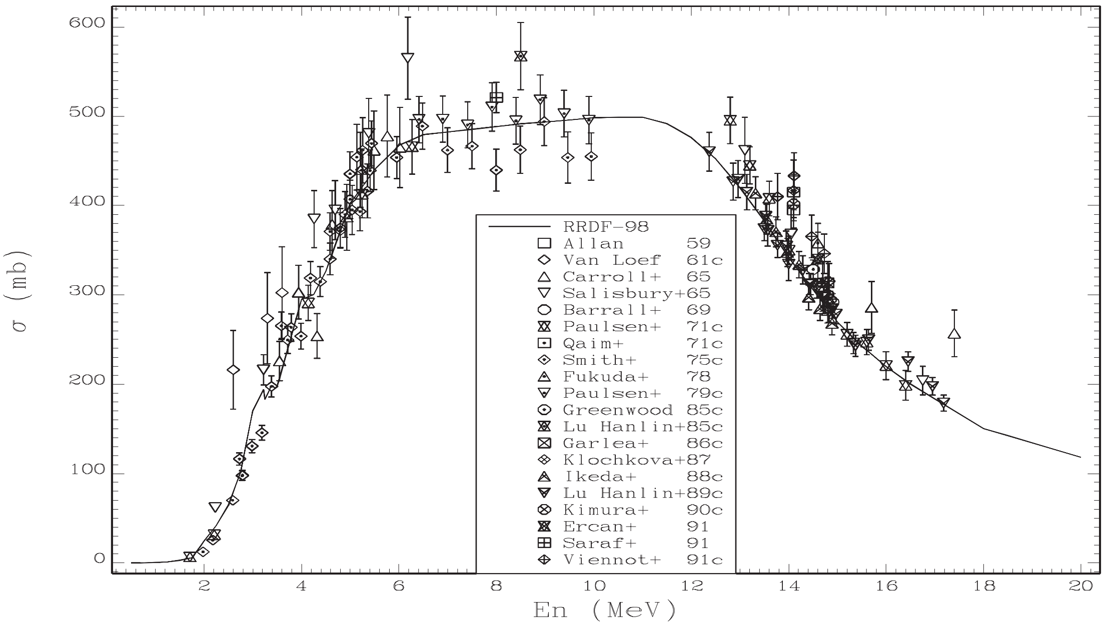

Fig. 7. Recommended ^{59}Fe(n,p)^{54}Mn reaction excitation function compared with experimental data

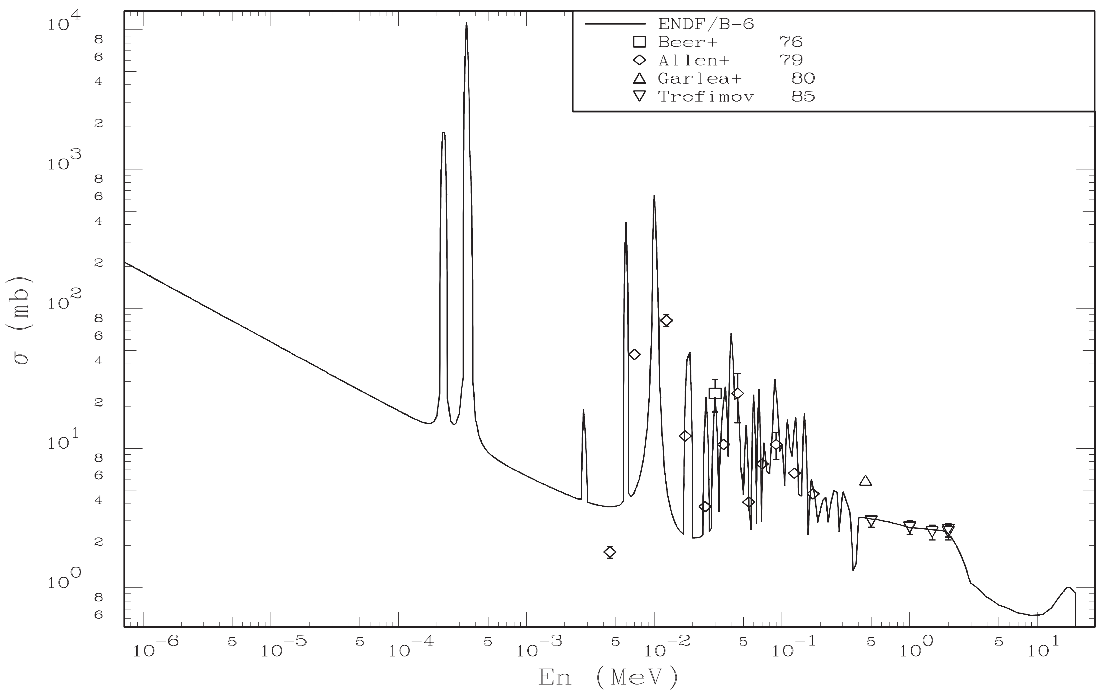

Fig. 8. Recommended ^{58}Fe(n,γ)^{59}Fe reaction excitation function compared with experimental data

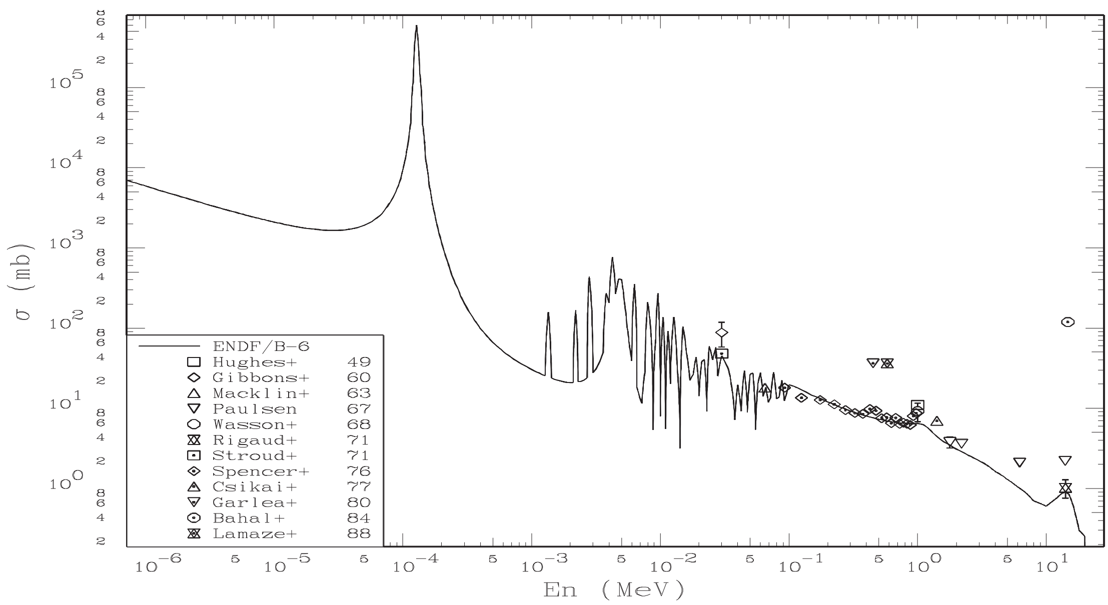

Fig. 9. Recommended $^{59}Co(n,\gamma)^{60}Co$ reaction excitation function compared with experimental data

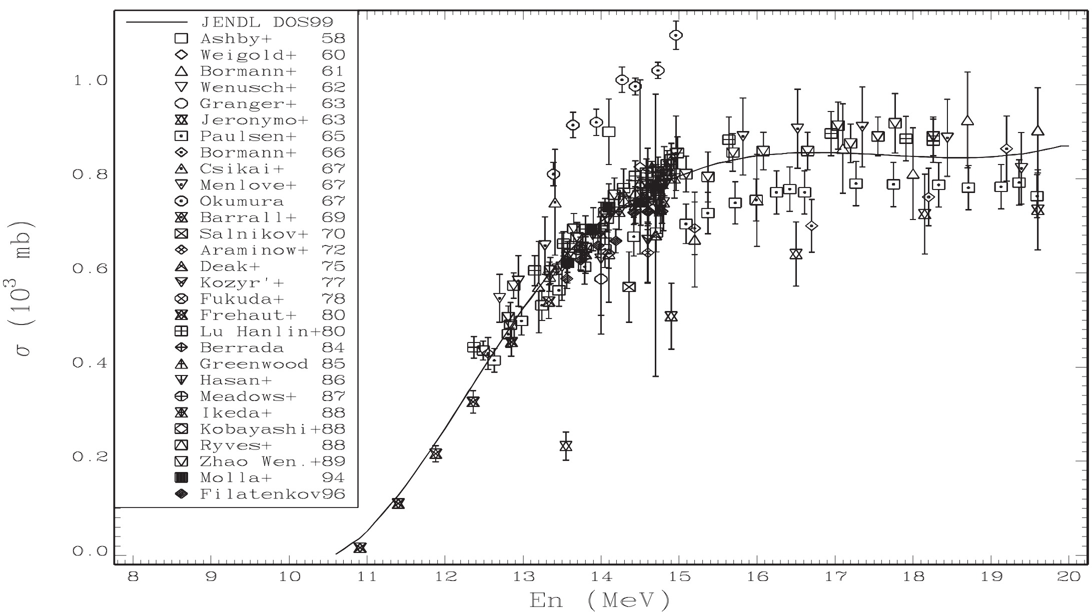

Fig. 10. Recommended $^{59}Co(n,2n)^{58m+g}Co$ reaction excitation function compared with experimental data

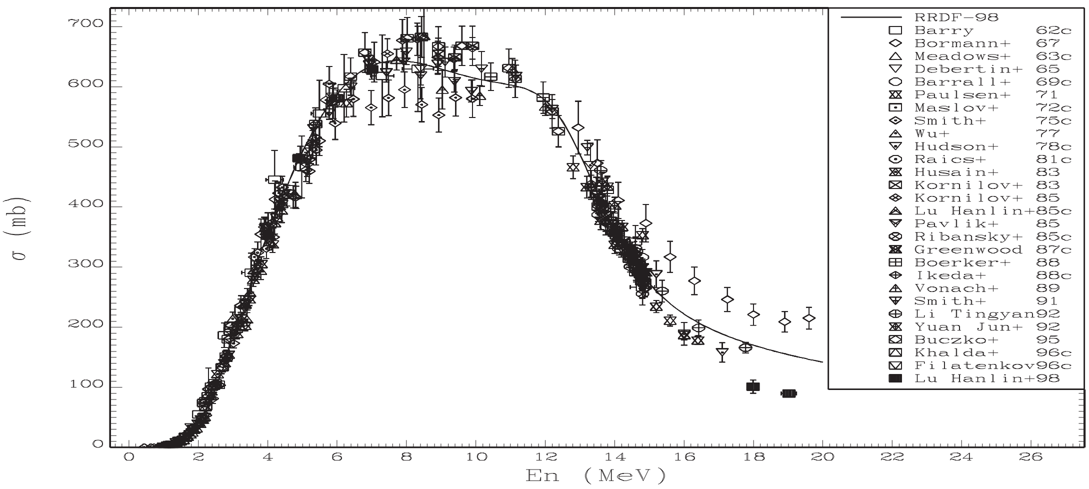

Fig. 11. Recommended $^{58}Ni(n,p)^{58m+g}Co$ reaction excitation function compared with experimental data

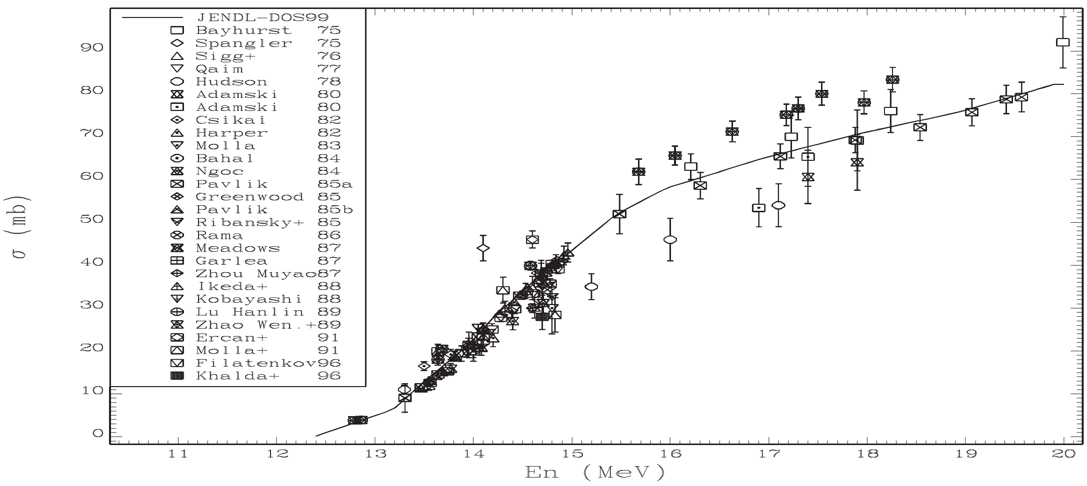

Fig. 12. Recommended $^{58}Ni(n,2n)^{57}Ni$ reaction excitation function compared with experimental data

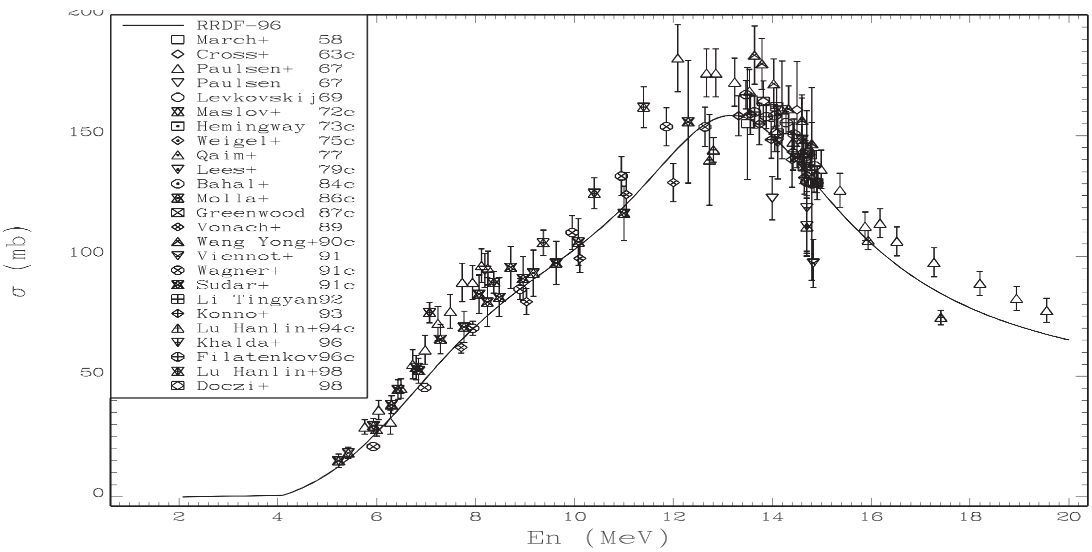

Fig. 13. Recommended ^{60}Ni(n,p)$^{60m+g}$Co reaction excitation function compared with experimental data

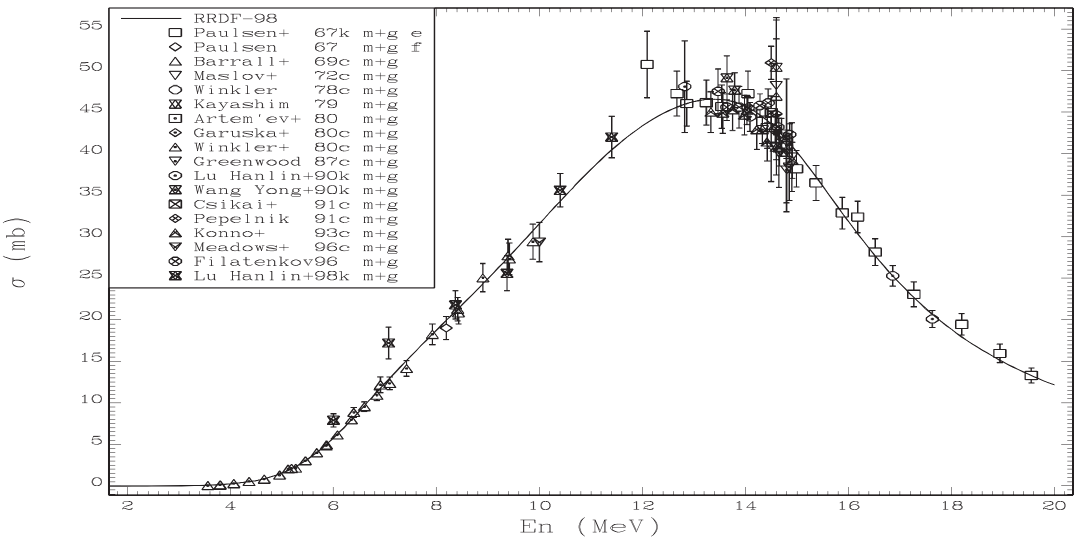

Fig. 14. Recommended ^{63}Cu(n,α)$^{60m+g}$Co reaction excitation function compared with experimental data

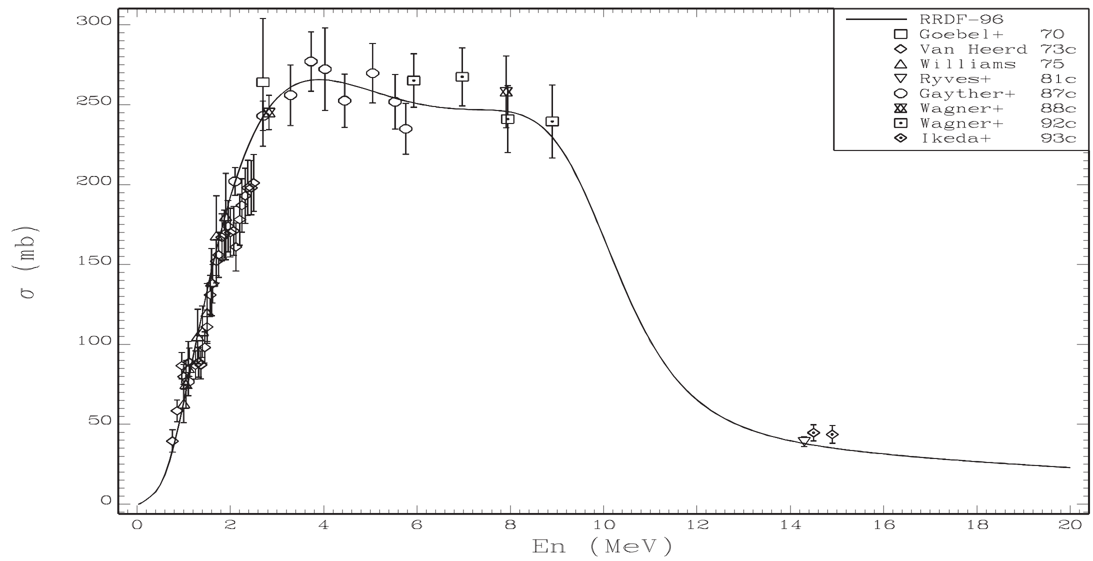

Fig. 15. Recommended ^{93}Nb(n,n')^{93m}Nb reaction excitation function compared with experimental data

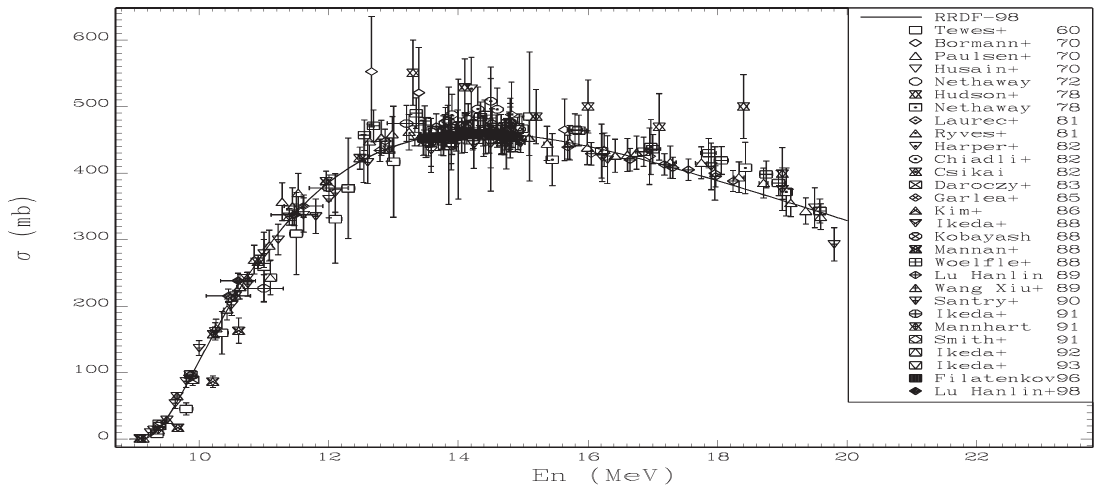

Fig. 16. Recommended ^{93}Nb(n,2n)^{92m}Nb reaction excitation function compared with experimental data

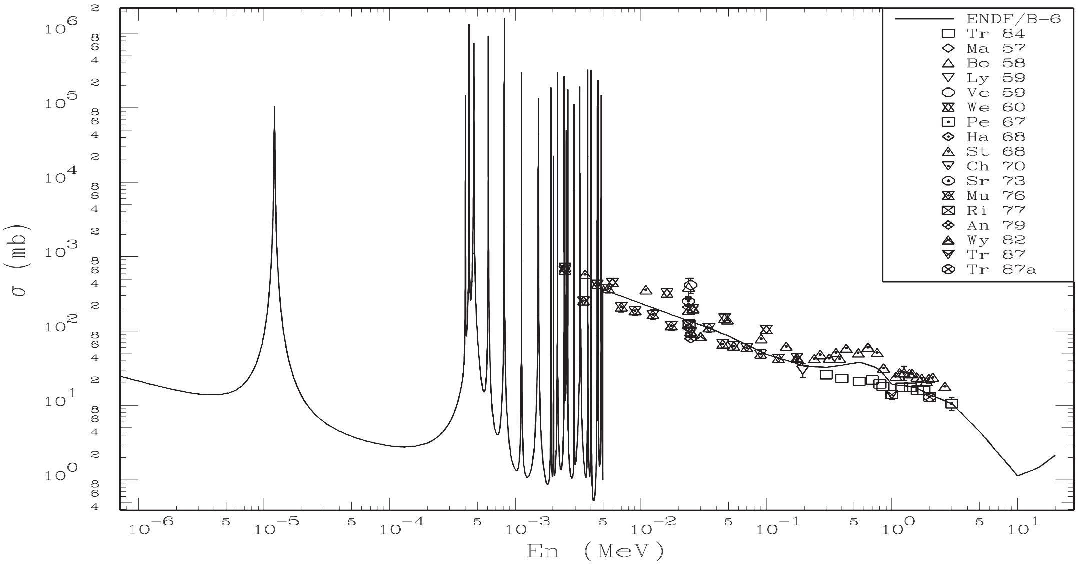

Fig. 17. Recommended $^{98}Mo(n,\gamma)^{99}Mo$ reaction excitation function compared with experimental data

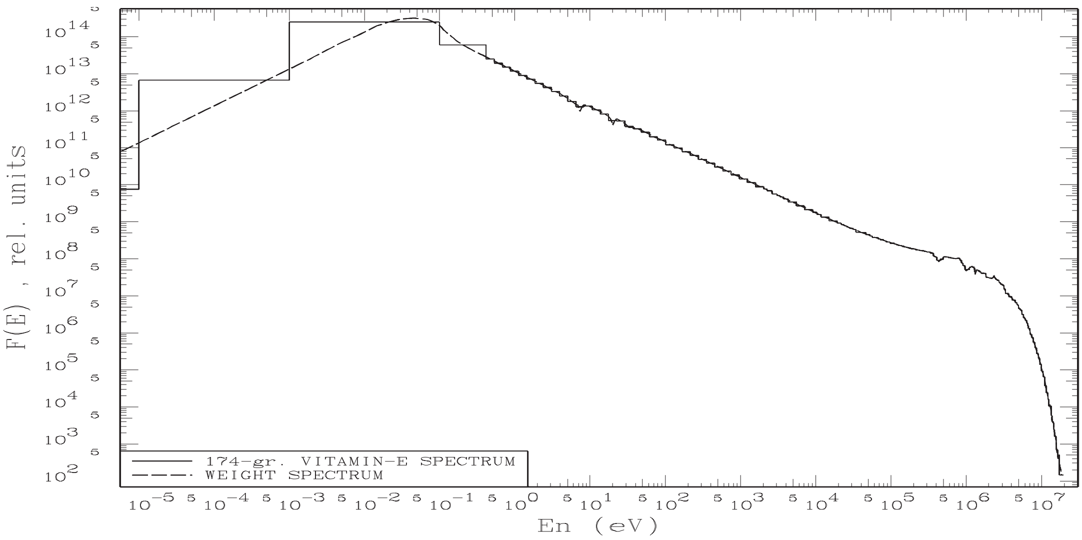

Fig. 18. Core averaged 174-group neutron spectrum and weighting function used for preparing the 47-group cross-sections

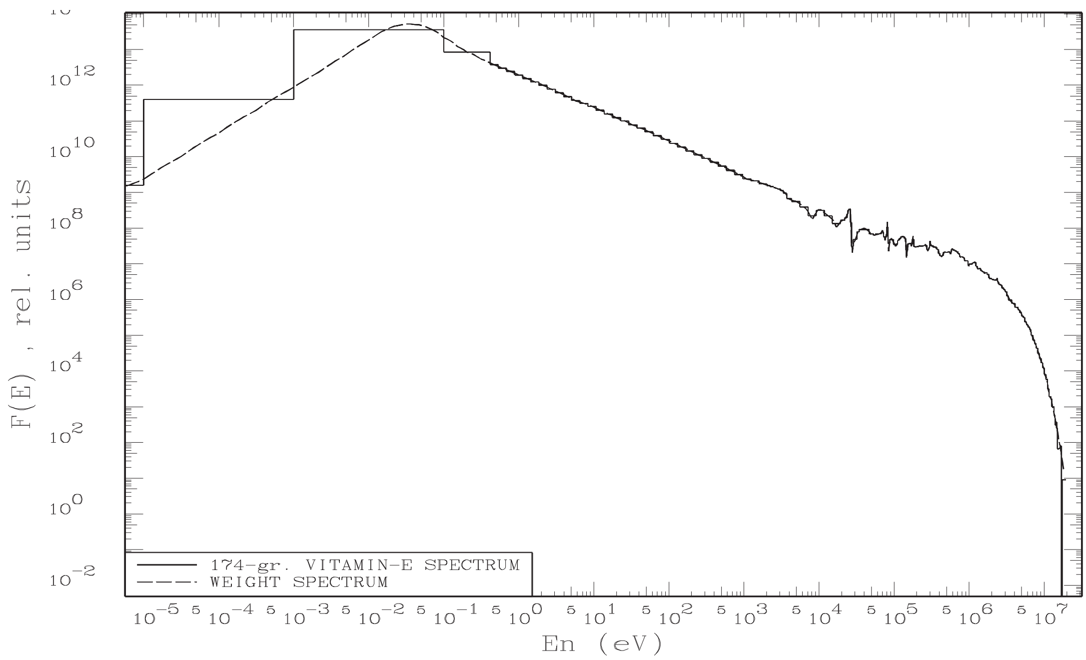

Fig. 19. 174-group neutron spectrum in core shroud and weighting function used for preparing the 47-group cross-sections

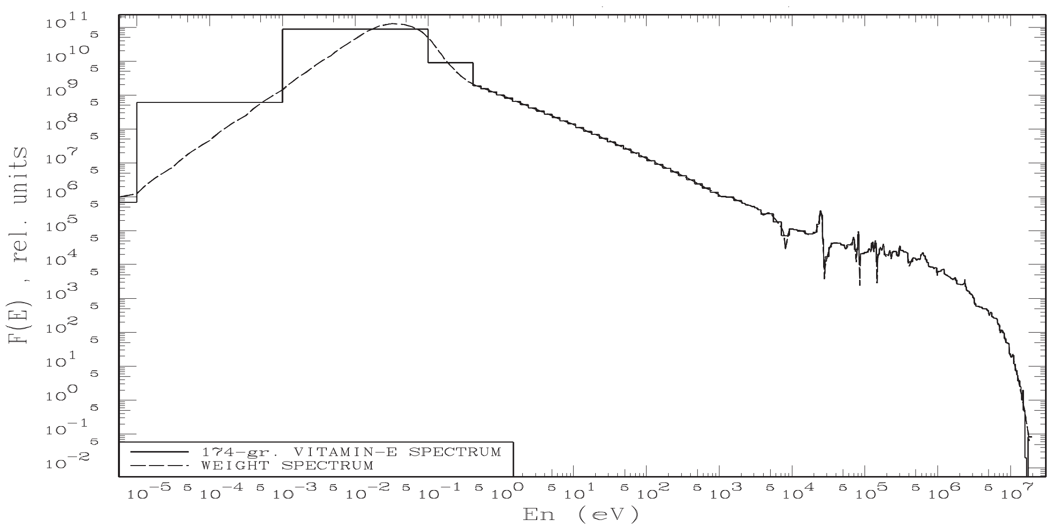

Fig. 20. 174-group neutron spectrum averaged over the volume of the first 4 cm of the vessel and weighting function used for preparing the 47-group cross-sections

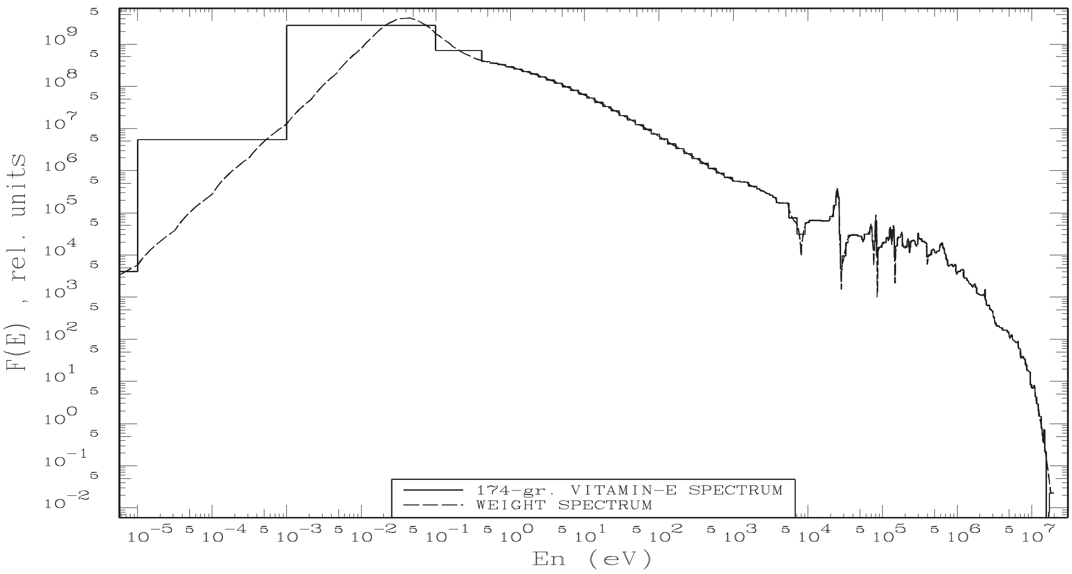

Fig. 21. 174-group neutron spectrum averaged over the 9 cm central vessel layer and weighting function used for preparing the 47-group cross-sections

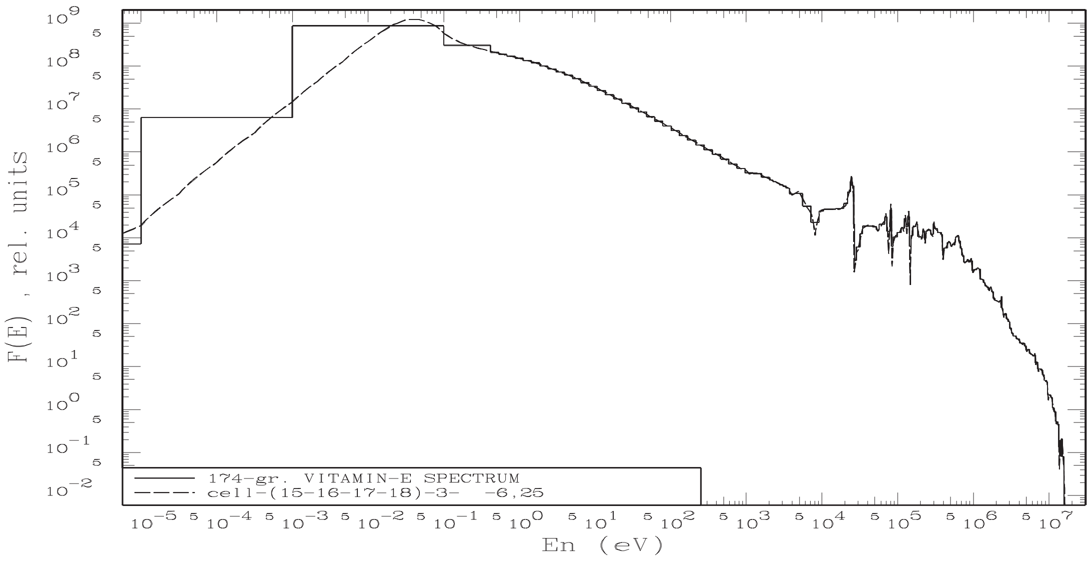

Fig. 22. 174-group neutron spectrum averaged over the 6.25 cm outer vessel layer and weighting function used for preparing the 47-group cross-sections

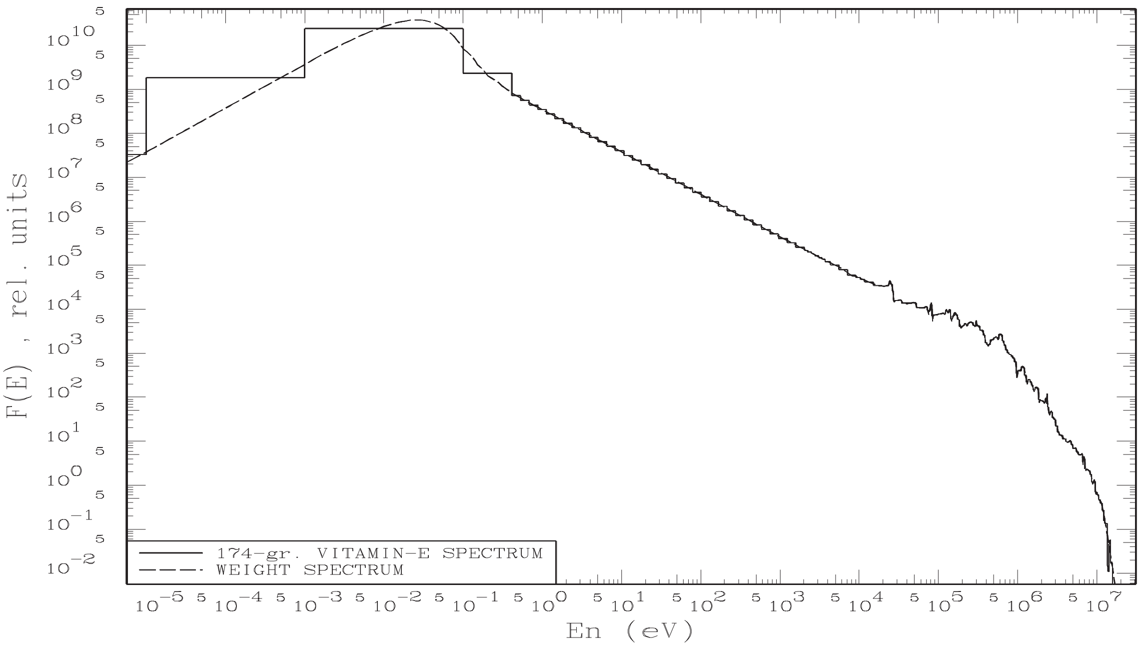

Fig. 23. 174-group neutron spectrum averaged over the first 12 cm concrete layer and weighting function used for preparing the 47-group cross-sections

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EVALUATION OF $^{54}Fe(n,2n)^{53m+g}Fe$ REACTION CROSS SECTIONS FOR HIGH ENERGY DOSIMETRY APPLICATIONS

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The new evaluation of excitation function for the high energy threshold $^{54}Fe(n,2n)^{53m+g}Fe$ dosimetry reaction in the energy range from the threshold to 20 MeV is briefly described. The cross section uncertainties and the covariance matrix were estimated simultaneously from the analysis. The adopted curve is compared to the available processed experimental data and the existing FEI-93, ENDF/B-VI and JENDL-3.2 evaluations. The ENDF-6 formatted data file is available from the Web site of the Russian Nuclear Data Center (RNDC) online (http://www.rndc.ippe.obninsk.ru).

Introduction

Several threshold neutron activation reactions/detectors have been proposed [1] to measure the higher than 16 MeV neutron flux including $^{54}Fe(n,2n)^{53m+g}Fe$ reaction as candidate. The accuracy requested is about 20%. The energy range is from threshold to 20 MeV. The lack of reliable evaluations for this reaction in national data bases stimulated effort to construct a new excitation function for inclusion to the Russian Reactor Dosimetry File (RRDF-98). This RRDF-98 evaluation supersedes previous preliminary FEI-93 evaluation performed by authors in 1993 [2] for the following reasons. First and foremost, in the course of previous evaluation [2], the cross section data for $^{27}Al(n,p)^{27}Mg$ monitor reaction have been taken according to the obsolete recommendation [3]. Consequently, it was necessary to make corrections of experimental data measured relative to $^{27}Al(n,p)^{27}Mg$ reaction. Furthermore, the previous evaluated curve FEI-93 did not take into account experimental data measured by Andreev&Serov [4]. Moreover, high precision measurements covering wide energy ranges from 15.2 to 18.3 MeV and from 16.07 to 20.36 MeV have been reported recently by Fessler [5].

Analysis and processing of experimental data

The threshold of $^{54}Fe(n,2n)^{53m+g}Fe$ is 13.626 MeV. In the energy range from threshold to 16.63 MeV $^{54}Fe(n,2n)$ reaction leads to the ^{53}Fe formation in the ground state. At neutron energies exceeding 16.63 MeV the 3.04 MeV isomer level of the ^{53}Fe ($J=19/2^-$, $T_{1/2}=2.58$ min) is excited. The transition from isomer level to ground state is realized with a probability of 100%.

The $^{54}Fe(n,2n)^{53m+g}Fe$ reaction cross-section measurements cover range from the threshold to 20.36 MeV. All used the activation technique. The list of examined experiments [4-28] for the $^{54}Fe(n,2n)^{53m+g}Fe$ reaction cross-section is given in Table 1. Columns 1, 2 contain the lower and upper limits of the energy range under consideration in the experiment and the number of experimental points within this range. Methods used for measurements of induced activity and neutron flux monitoring are described in columns 3, 4. The columns 5 and 6 list the names of the first author of the publication and reference number. The original experimental data are shown in Fig. 1.

The original experimental data have been renormalized, if necessary, to the up-to-dated recommended standard cross-sections [3,29,30], quantum and positron yields [31]. Cross section data for the monitor reaction $^{27}Al(n,p)^{27}Mg$ were taken from the recent re-evaluation [32]. This new re-evaluation gives the cross section data in the energy range 13-16 MeV with accuracy from 1.34 to 1.8%. The changes in half-lives of radioactive nuclei have been taken into account by means of the increasing of cross-section errors.

The careful inspection of available experimental data disclosed that in the energy range from threshold to 15 MeV results of recent Greenwood's and Viennot's experiments [24,27] are consistent with each other. The measurements of Katoh et al. [26] do not contradict to Greenwood's and Viennot's data but exceed them below 14.7 MeV. The measurements of Qaim [17] and Bahal & Smither [23] at energy 14.7 MeV practically coincide. They are also in agreement with results of experiments described above. The same is true for measurements reported by Chittenden et al. at 14.8 MeV [10] and Depraz et al. at 15.0 MeV [8]. In spite of corrections the results of many experiments in the energy range from threshold to 15 MeV [6,9,11,13,14,18,20,25] are inconsistent with both theoretical model calculations and measurements considered above. In particular, cross-sections measured in those experiments are higher by a factor of 2 to 4 than Greenwood's and Viennot's data. By this reason they have been disregarded.

Terrel et al. [7] data measured at 16.89 and 17.89 MeV have been corrected introducing the F_c=0.5 coefficient, because authors used double decreased value for positron yield. Andreev & Serov performed two sets of measurements to obtain the cross section data for $^{54}Fe(n,2n)^{53m+g}Fe$ reaction for neutron energies from 13.87 to 17.4 MeV [4]. Both these two sets of absolute values of data are diverged from other data being strongly large. Analysis of Andreev's experimental data shows that relative cross section experimental curve looks rather well. Therefore, Andreev's data have been renormalized using preliminary evaluated integral cross section value in the neutron energy interval from 15.0 to 16.5 MeV. Renormalized experimental data by Andreev & Serov are well consistent with measurement results as well as with new experimental data for $^{54}Fe(n,2n)^{53m+g}Fe$ reaction measured recently by Fessler [5] at Geel (see Fig. 2.). Relative behavior of $^{54}Fe(n,2n)^{53m+g}Fe$ excitation function measured by Bormann et al. [20] in energy range from 14.05 to 18.23 MeV contradicts to most of other data sets as well as to theoretical calculation results. Therefore, Bormann et al. experimental data [20] have not been used for evaluation of excitation function for $^{54}Fe(n,2n)^{53m+g}Fe$ reaction. The resulting experimental data for the $^{54}Fe(n,2n)^{53m+g}Fe$ reaction cross-section are given in Table 2 and shown in Fig. 2. Table 2 includes information about average energy and energy spread of incident neutrons, the uncertainty of average neutron energy, cross-sections values and their uncertainties given by authors, corrected cross-sections and their uncertainties. For all experimental sets included in data bases the analysis of cross section error components has been performed and average correlation coefficient for each experimental data set has been calculated (see Table 3). In doing so, according to the information reported by authors, two independent sets (A and B) of measurements performed by Andreev & Serov [4], Ryves et al. [21.22] and Fessler [5] have been used to calculate the average correlation coefficients for each data set.

Table 1.

SUMMARY OF EXPERIMENTS FOR THE REACTION ^{54}Fe (n,2n)$^{53m+g}$ Fe

Energy-range [MeV]		Nr. of data points	Method of measurement	Monitor	References		Ref.
14.10	14.10	1	Act, Prop. Counter, Beta+	Cu63(n,2n)Cu62	Allan	56	[6]
16.89	17.89	2	Act, GEMUC, Beta-	Fe56(n,p)Mn56 norm at 14.30 MeV	Terrell+	58	[7]
15.00	15.00	1	Act, Prop. counter, Beta	Cu63(n,2n)Cu62	Depraz+	60	[8]
14.40	14.40	1	Act, Two NaI(Tl), Ann.Gammas coinc.	Cu63(n,2n)Cu62	Rayburn	61	[9]
14.80	14.80	1	Act, B+	Al27(n,a)Na24 and Cu63(n,2n)Cu62	Chittenden+	61	[10]
14.10	14.10	1	Act,Boric acid counter+NaI(Tl),Gamma	NO INFORMATION GIVEN	Pollehn+	61	[11]
14.50	14.50	1	Activation method	Al27(n,a)Na24	Cross+	63	[12]
14.10	14.10	1	Act, Two NaI(Tl) ,Ann.Gammas Coinc.	Cu63(n,2n)Cu62	Carles	63	[13]
14.70	14.70	1	Act, NaI(Tl), Gamma	Cu63(n,2n)Cu62	Strain+	65	[14]
14.05	16.75	2	Act., NaI(Tl) det., Ann. Gamma	NO INFORMATION GIVEN	Salisbury+	65	[15]
14.60	14.60	1	Act., NaI(Tl) det., Gamma	Cu63(n,2n)Cu62	Csikai	65	[16]
13.87	17.40	10	Act, Solid Scint., Ann. Gamma	Cu63(n,2n)Cu62 norm at 14.30 MeV	Andreev+	68	[4]
13.87	17.40	10	Act, Solid Scint., Ann.Gamma coinc.	Cu63(n,2n)Cu62 norm at 14.30 MeV	Andreev+	68	[4]
14.70	14.70	1	Act. method, Ge(Li) det., Ann.Gamma	Al27(n,p)Mg27	Qaim	72	[17]
14.60	14.60	1	Act., NaI(Tl) det., Gamma	Cu63(n,2n)Cu62	Araminowicz+	73	[18]
14.80	14.80	1	Act. method, Ge(Li) det., Ann.Gamma	Al27(n,p)Mg27	Sigg+	75	[19]
14.05	18.23	11	Act, NaI, Gamma and G-G coincidence	1-H-1(n,n)1-H-1 norm at 14.30 MeV	Bormann+	76	[20]
15.30	18.95	6	Act, 4PI Beta-Gamma coinc. counter	1-H-1(n,n)1-H-1 norm at 14.77 MeV	Ryves+	78	[21]
14.65	19.00	6	Act, 4PI Beta-Gamma coinc. counter	Fe56(n,p)Mn56	Ryves+	78	[22]
14.70	14.70	1	Act. method, Ge(Li) det., Gamma	Al27(n,p)Mg27	Bahal+	84	[23]
13.95	14.64	6	Act, Ge(Li), Gamma	Al27(n,p)Mg27 and Al27(n,a)Na24	Greenwood+	85	[24]
14.60	14.60	1	Act. method, Ge(Li) det., Gamma	T(d,n)He4 assoc.pt. norm at 14.40 MeV	Zhou Muyao+	87	[25]
13.70	14.87	5	Act., HPGe, Gamma	Al27(n,p)Mg27	Katoh+	89	[26]
13.93	14.83	5	Act. method, Ge(Li) det., Gamma	Al27(n,p)Mg27	Viennot+	91	[27]
14.60	14.60	1	Act. method, HP Ge detector, Gammas	Al27(n,a)Na24	Ercan+	91	[28]
15.22	18.30	7	Act. method, HP Ge detector, Gammas	Al27(n,p)Mg27 + 2 monit. reactions	Fessler+	98	[5]
16.07	20.36	5	Act. method, HP Ge detector, Gammas	Al27(n,p)Mg27 + 2 monit. reactions	Fessler+	98	[5]

Table 2.

CROSS SECTION DATA USED FOR THE EVALUATION of $^{54}Fe(n,2n)^{53m+g}Fe$ EXCITATION FUNCTION

NR.	E-NEUTR [MeV]	ERR.CENTR [MeV]	WIDTH [MeV]	SIGMA (ORIG) [mb]	ERROR (ORIG) [mb]	CORR.APPL.	SIGMA (CORR) [mb]	ERROR (CORR) [mb]	REFERENCE	
1	13.700	0.050	0.000	0.220	0.198	1 5	0.215	0.215	Katoh+	89
2	13.870	0.098	0.490	12.800	6.501	5 6 8	6.088	3.107	Andreev+	68
3	13.870	0.098	0.490	1.300	7.400	5 6 8	0.673	3.831	Andreev+	68
4	13.930	0.030	0.120	0.800	0.800	1 5	0.780	0.780	Viennot+	91
5	13.950	0.070	0.200	0.680	0.095	1 3 5	0.633	0.227	Greenwood+	85
6	14.010	0.070	0.200	0.800	0.144	1 3 5	0.742	0.286	Greenwood+	85
7	14.010	0.150	0.000	2.100	0.441	1 5	2.044	1.177	Katoh+	89
8	14.050	0.210	1.050	2.830	0.050	4 8	2.830	1.920	Salisbury+	65
9	14.300	0.104	0.520	9.300	7.300	5 6 8	4.424	3.480	Andreev+	68
10	14.300	0.104	0.520	15.100	9.901	5 6 8	7.817	5.140	Andreev+	68
11	14.300	0.035	0.140	2.100	0.900	1 5	2.009	0.884	Viennot+	91
12	14.310	0.070	0.200	2.550	0.153	1 3 5	2.312	0.677	Greenwood+	85
13	14.350	0.150	0.000	5.300	1.060	1 5	5.114	1.994	Katoh+	89
14	14.460	0.070	0.200	3.700	0.111	1 3 5	3.319	0.903	Greenwood+	85
15	14.470	0.043	0.170	3.600	1.000	1 5	3.403	0.982	Viennot+	91
16	14.500	0.100	0.000	8.500	0.801	1 5	8.566	2.162	Cross+	63
17	14.600	0.000	0.000	9.000	2.001	1	9.032	2.008	Ercan+	91
18	14.610	0.070	0.200	5.050	0.303	1 3 5	4.484	1.260	Greenwood+	85
19	14.640	0.070	0.200	5.850	0.175	1 3 5	5.184	1.400	Greenwood+	85
20	14.640	0.150	0.000	9.000	1.800	1 5	8.648	3.329	Katoh+	89
21	14.650	0.100	0.100	11.900	0.300	1 58	11.864	2.592	Ryves+	78
22	14.700	0.030	0.300	8.000	1.600	1 4 8	7.429	1.399	Qaim	72
23	14.700	0.040	0.000	7.900	0.700	1 3 6 8	7.391	1.653	Bahal+	84
24	14.730	0.058	0.230	7.100	1.801	1 5	6.585	1.830	Viennot+	91
25	14.800	0.090	0.900	7.900	0.800	1 5	7.901	1.700	Chittenden+	61
26	14.800	0.060	0.200	15.000	8.670	1 3 5	12.681	7.770	Sigg+	75
27	14.830	0.062	0.250	8.100	1.801	1 5	7.466	1.908	Viennot+	91
28	14.870	0.150	0.000	10.900	2.180	1 5	10.452	3.627	Katoh+	89
29	15.000	0.040	0.400	7.000	7.000	1	7.376	7.369	Depraz+	60

CONTINUE OF TABLE 2

NR.	E-NEUTR [MeV]	ERR.CENTR [MeV]	WIDTH [MeV]	SIGMA (ORIG) [mb]	ERROR (ORIG) [mb]	CORR. APPL.	SIGMA (CORR) [mb]	ERROR (CORR) [mb]	REFERENCE	
30	15.100	0.114	0.570	36.900	9.302	5 6 8	17.551	4.511	Andreev+	68
31	15.100	0.114	0.570	19.500	10.003	5 6 8	10.095	5.203	Andreev+	68
32	15.220	0.030	0.220	22.600	2.000	0	22.600	2.000	Fessler+	98
33	15.300	0.200	0.300	24.000	2.302	1 5 8	23.876	6.055	Ryves+	78
34	15.470	0.026	0.130	57.500	9.901	5 6 8	27.350	4.904	Andreev+	68
35	15.470	0.026	0.130	50.000	12.700	5 6 8	25.885	6.702	Andreev+	68
36	15.640	0.030	0.270	26.600	2.000	0	26.600	2.000	Fessler+	98
37	15.820	0.058	0.290	56.400	9.701	5 6 8	26.827	4.805	Andreev+	68
38	15.820	0.058	0.290	60.000	11.400	5 6 8	31.061	6.103	Andreev+	68
39	16.000	0.034	0.170	59.000	9.304	5 6 8	28.063	4.642	Andreev+	68
40	16.000	0.034	0.170	54.100	10.604	5 6 8	28.007	5.666	Andreev+	68
41	16.040	0.036	0.180	75.000	10.800	5 6 8	35.674	5.437	Andreev+	68
42	16.040	0.036	0.180	74.600	12.607	5 6 8	38.620	6.809	Andreev+	68
43	16.060	0.300	0.300	44.700	2.302	1 5 8	42.713	8.188	Ryves+	78
44	16.070	0.020	0.200	39.100	3.000	0	39.100	3.000	Fessler+	98
45	16.510	0.250	0.250	56.000	3.702	1 5 8	53.759	7.096	Ryves+	78
46	16.510	0.030	0.300	48.200	3.700	0	48.200	3.700	Fessler+	98
47	16.530	0.050	0.250	95.800	11.506	5 6 8	45.567	5.928	Andreev+	68
48	16.530	0.050	0.250	94.200	14.902	5 6 8	48.766	8.090	Andreev+	68
49	16.550	0.100	0.240	58.000	12.400	1 5 8	57.700	12.619	Ryves+	78
50	16.750	0.110	0.550	50.400	5.000	4 8	50.400	6.200	Salisbury+	65
51	16.890	0.160	0.320	120.000	36.000	1 6	60.000	16.980	Terrell+	58
52	16.980	0.056	0.280	97.400	12.506	5 6 8	46.328	6.384	Andreev+	68
53	16.980	0.056	0.280	116.000	16.820	5 6 8	60.052	9.212	Andreev+	68
54	17.000	0.100	0.220	49.000	6.502	1 5 8	48.747	6.639	Ryves+	78
55	17.060	0.020	0.220	61.100	4.600	0	61.100	4.600	Fessler+	98
56	17.350	0.200	0.200	73.100	2.902	1 5 8	71.928	6.308	Ryves+	78
57	17.370	0.100	0.220	65.000	6.006	1 5 8	64.664	6.259	Ryves+	78
58	17.400	0.060	0.300	132.300	14.103	5 6 8	62.928	7.407	Andreev+	68
59	17.400	0.060	0.300	130.100	18.734	5 6 8	67.351	10.264	Andreev+	68
60	17.540	0.030	0.290	63.100	5.200	0	63.100	5.200	Fessler+	98

NR.	E-NEUTR [MeV]	ERR.CENTR [MeV]	WIDTH [MeV]	SIGMA (ORIG) [mb]	ERROR (ORIG) [mb]	CORR.APPL.	SIGMA (CORR) [mb]	ERROR (CORR) [mb]	REFERENCE
61	17.820	0.030	0.250	71.400	6.200	0	71.400	6.200	Fessler+ 98
62	17.860	0.100	0.210	86.000	9.107	1 5 8	85.556	9.112	Ryves+ 78
63	17.890	0.040	0.080	170.000	51.000	1 6	85.000	24.055	Terrell+ 58
64	17.910	0.030	0.300	69.400	6.900	0	69.400	6.900	Fessler+ 98
65	18.060	0.190	0.190	87.100	4.303	1 5 8	85.615	5.625	Ryves+ 78
66	18.210	0.040	0.400	72.600	4.800	0	72.600	4.800	Fessler+ 98
67	18.300	0.040	0.320	75.600	4.900	0	75.600	4.900	Fessler+ 98
68	18.950	0.100	0.210	85.000	5.907	1 5 8	84.561	5.936	Ryves+ 78
69	19.000	0.190	0.190	98.700	5.902	1 5 8	98.719	5.726	Ryves+ 78
70	19.150	0.030	0.300	81.400	7.700	0	81.400	7.700	Fessler+ 98
71	20.360	0.040	0.350	88.200	7.200	0	88.200	7.200	Fessler+ 98

Correction codes:
0) No correction applied.
1) Cross-section renormalized to the new recommended values of reference cross-section used in measurement.
2) Uncertainty in the reference cross-section data is included in the total cross-section data error.
3) Cross-section renormalized to the new recommended decay data (half-life, emission probability etc.)
4) Uncertainty in the decay data is included in the total cross-section data error.
5) Error given in publication did not include some partial errors. See text for details.
6) Special correction. See text for details.
7) The center of energy resolution function was shifted. See text for details.
8) Uncertainty in neutron energy is included in the total cross-section data error.
9) Total uncertainty in the cross-section data was evaluated.

Table 3.

Average correlation coefficients (F-corr) for the experimental data used for the evaluation of the $^{54}Fe(n,2n)^{53m+g}$ reaction excitation function

Experimental data			F-corr	Experimental data			F-corr
1	Terrell+	58	0.24	10	Ryves+	78A	0.22
2	Depraz+	60	0.00	11	Ryves+	78B	0.61
3	Chittenden+	61	0.00	12	Bahal+	84	0.00
4	Cross+	63	0.00	13	Greenwood+	85	0.80
5	Salisbury+	65	0.30	14	Katoh+	89	0.69
6	Andreev+	68A	0.50	15	Viennot+	91	0.37
7	Andreev+	68B	0.44	16	Ercan+	91	0.00
8	Qaim	72	0.00	17	Fessler+	98A	0.29
9	Sigg+	75	0.00	18	Fessler+	98B	0.28

Evaluated $^{54}Fe(n,2n)^{53m+g}Fe$ reaction excitation function

After considering and processing all experimental data collected in Table 1 a smooth curve was drawn through these data points (given in Table 2) as calculated using the generalized least square method [33]. The adopted curve is compared to the available processed measurements of Table 2 from the threshold to 15 MeV in Fig. 3 and to 20 MeV in Fig. 4.

As shown in Figs. 5 and 6 the existing ENDF/B-VI and JENDL-3.2 evaluations diverge from the adopted RRDF-98 curve. Both run higher on 10-12% in the important for dosimetry energy range from threshold to 15.5 MeV (Fig. 5). The JENDL-3.2 curve has a rather strange shape strongly deviating from the experimental data (30-35%) and from both RRDF-98 and ENDF/B-VI evaluations in the energy range from 16.5 to 20 MeV (Fig. 6). Fig. 7 shows comparison of the preliminary FEI-93 curve and adopted RRDF-98 excitation function in comparison with processed experimental data.

The evaluated group cross-sections and their correlations are given in the Tables 4 and 5. Uncertainties given in the Table 4 were calculated for confidence level P=0.95 (2σ).

Table 4.

Group cross-sections and their uncertainties for the reaction $^{54}Fe(n,2n)^{53m+g}Fe$

ENERGY GROUP [MeV] to [MeV]	GROUP NUMBER	CROSS-SECTION [mb]	ERROR [mb]	ERROR [%]
13.700 - 13.900	1	0.41	0.21	52.28
13.900 - 14.100	2	1.11	0.32	28.84
14.100 - 14.300	3	2.30	0.46	20.00
14.300 - 14.500	4	4.02	0.61	15.12
14.500 - 14.700	5	6.28	0.77	12.33
14.700 - 14.900	6	9.08	0.96	10.60
14.900 - 15.100	7	12.41	1.16	9.38
15.100 - 15.300	8	16.21	1.36	8.40
15.300 - 15.500	9	20.43	1.55	7.58
15.500 - 16.000	10	28.60	1.84	6.42
16.000 - 16.500	11	40.90	2.27	5.55
16.500 - 17.000	12	52.77	2.72	5.15
17.000 - 17.500	13	63.19	3.04	4.80
17.500 - 18.000	14	71.69	3.20	4.47
18.000 - 19.000	15	80.69	3.65	4.52
19.000 - 20.000	16	87.81	5.51	6.27
20.000 - 21.000	17	91.15	8.06	8.84

Table 5.

Correlation matrix of group cross-sections for the $^{54}Fe(n,2n)^{53m+g}Fe$ reaction
Correlations are given in percentages

ENERGY GROUP [MeV] to [MeV]	GROUP NUMBER	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
13.700 - 13.900	1	100																
13.900 - 14.100	2	19	100															
14.100 - 14.300	3	25	11	100														
14.300 - 14.500	4	28	67	82	100													
14.500 - 14.700	5	30	61	78	89	100												
14.700 - 14.900	6	30	53	70	84	92	100											
14.900 - 15.100	7	29	44	61	77	89	95	100										
15.100 - 15.300	8	27	36	52	69	83	92	96	100									
15.300 - 15.500	9	26	30	44	61	76	86	93	97	100								
15.500 - 16.000	10	22	21	32	47	61	73	82	89	95	100							
16.000 - 16.500	11	15	12	17	26	37	47	58	67	77	91	100						
16.500 - 17.000	12	9	8	9	12	18	25	33	43	54	74	74	100					
17.000 - 17.500	13	5	8	7	6	8	12	18	26	37	57	82	96	100				
17.500 - 18.000	14	5	10	8	7	7	9	13	19	27	44	68	85	95	100			
18.000 - 19.000	15	6	10	13	15	16	18	19	21	23	29	40	52	67	85	100		
19.000 - 20.000	16	9	7	14	22	27	30	30	27	24	16	7	8	20	43	84	100	
20.000 - 21.000	17	9	4	13	24	31	35	35	32	26	12	-6	-13	-4	18	65	96	100

Conclusion

As expected and illustrated in Table 4, the greatest uncertainties of evaluated cross sections for the $^{54}Fe(n,2n)^{53m+g}Fe$ reaction were estimated for the nearthreshold energy range 13.7-14.5 MeV and varies from about 50 to 15%. Because of the importance of $^{54}Fe(n,2n)^{53m+g}Fe$ reaction for high energy dosimetry applications, we feel that additional precise measurements from 13.7 to 14.5 MeV are required.

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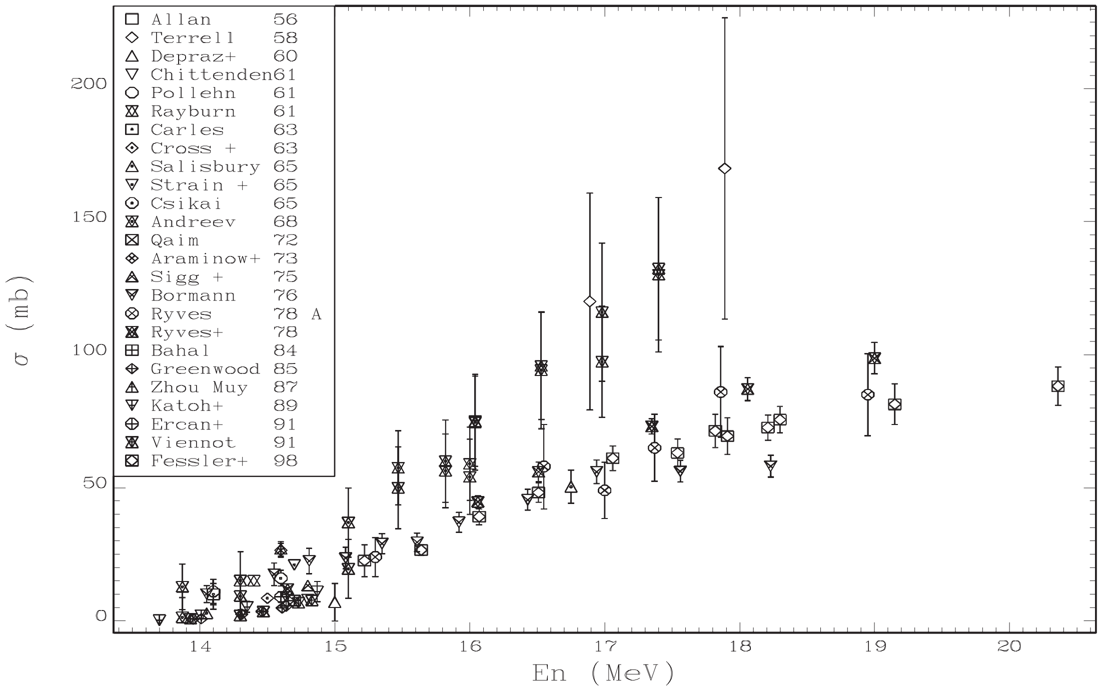

Fig. 1. The original experimental data for the $^{54}Fe(n,2n)^{53g+m}Fe$ reaction excitation function in the energy range from threshold to 20 MeV.

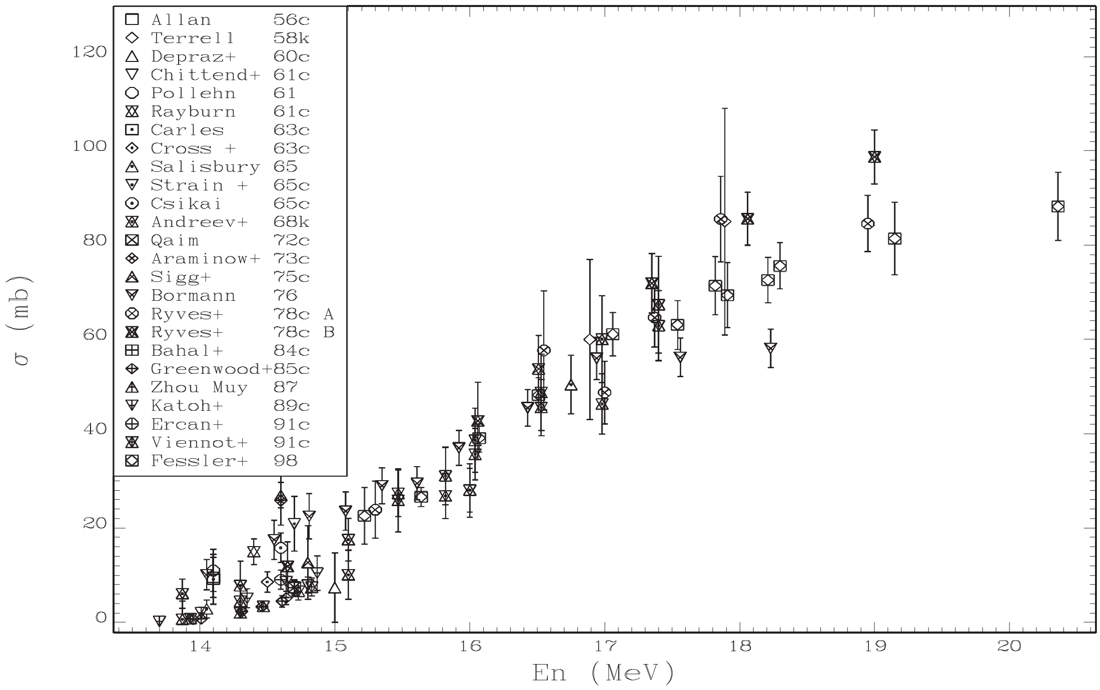

Fig. 2. The corrected experimental data for the $^{54}Fe(n,2n)^{53g+m}Fe$ reaction excitation function in the energy range from threshold to 20 MeV.

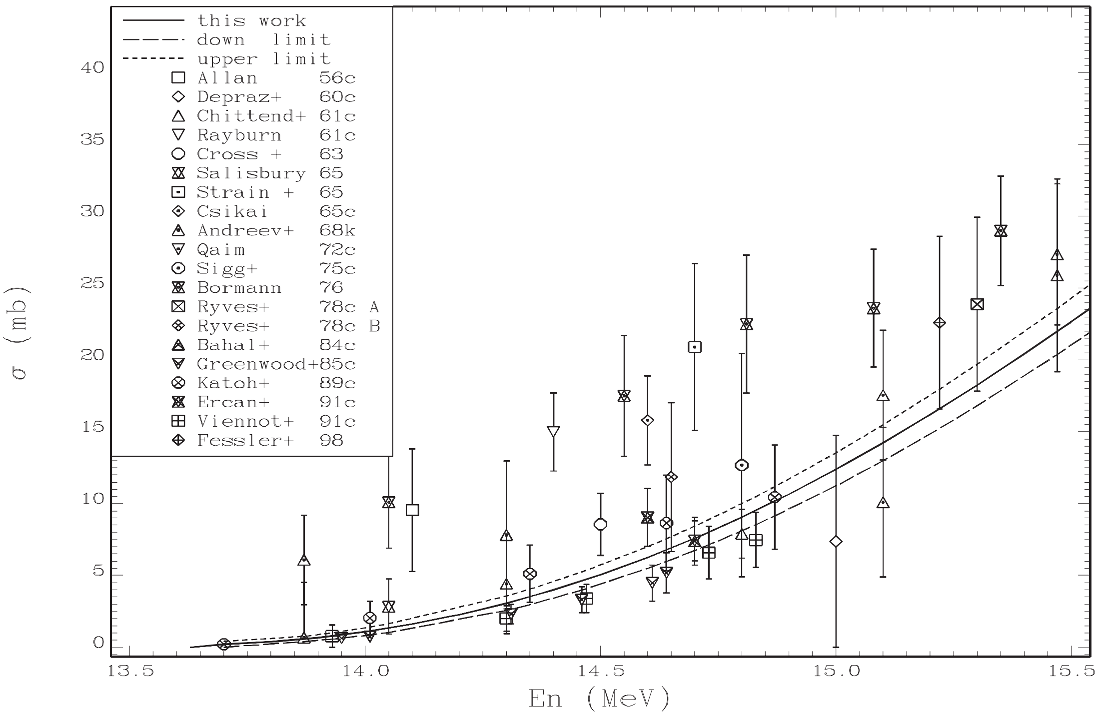

Fig. 3. The results of cross-section evaluation for the reaction $^{54}Fe(n,2n)^{53m+g}Fe$ from this work in the energy range from threshold to 15 MeV (dashed lines display 1 standard deviation error of evaluation).

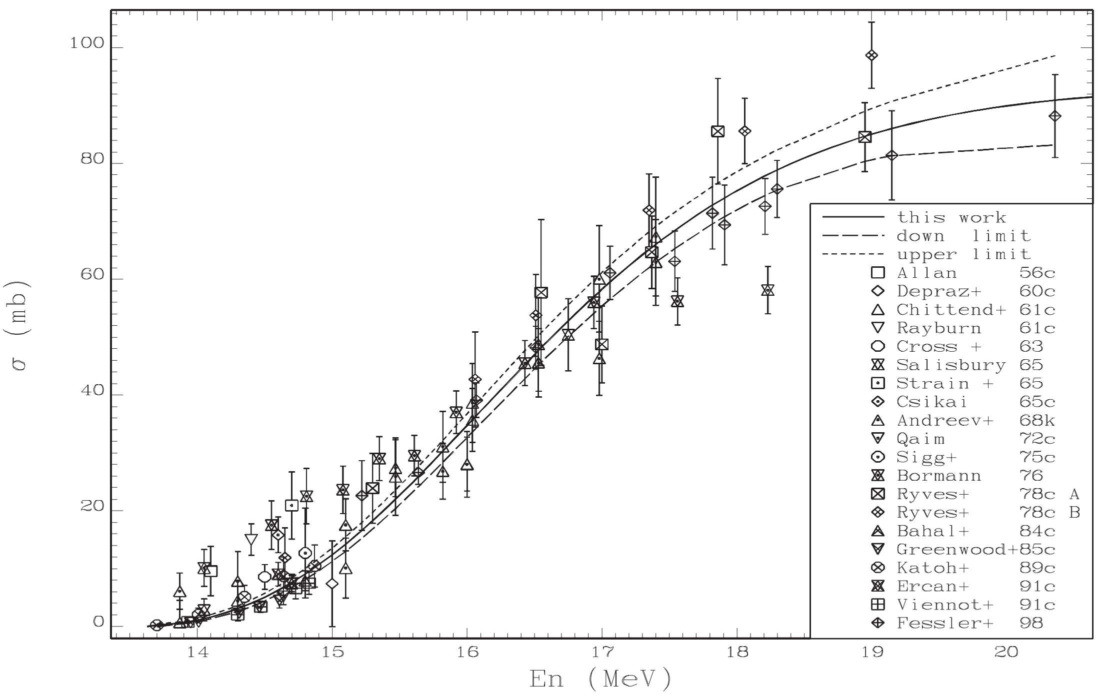

Fig. 4. The results of cross-section evaluation for the reaction $^{54}Fe(n,2n)^{53m+g}Fe$ from this work in the energy range from threshold to 20 MeV (dashed lines display 1 standard deviation error of evaluation).

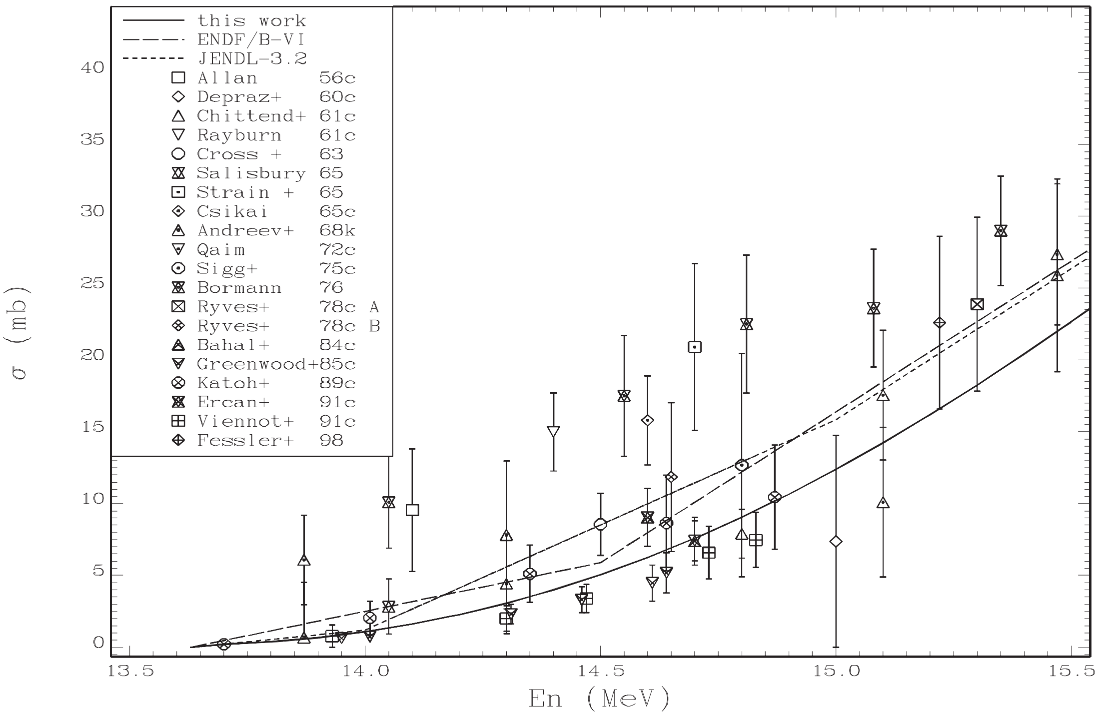

Fig. 5. The evaluated cross-section for the reaction $^{54}Fe(n,2n)^{53m+g}Fe$ in comparison with the experimental data and the JENDL-3 and ENDF/B-VI evaluations in the energy range from threshold to 15 MeV.

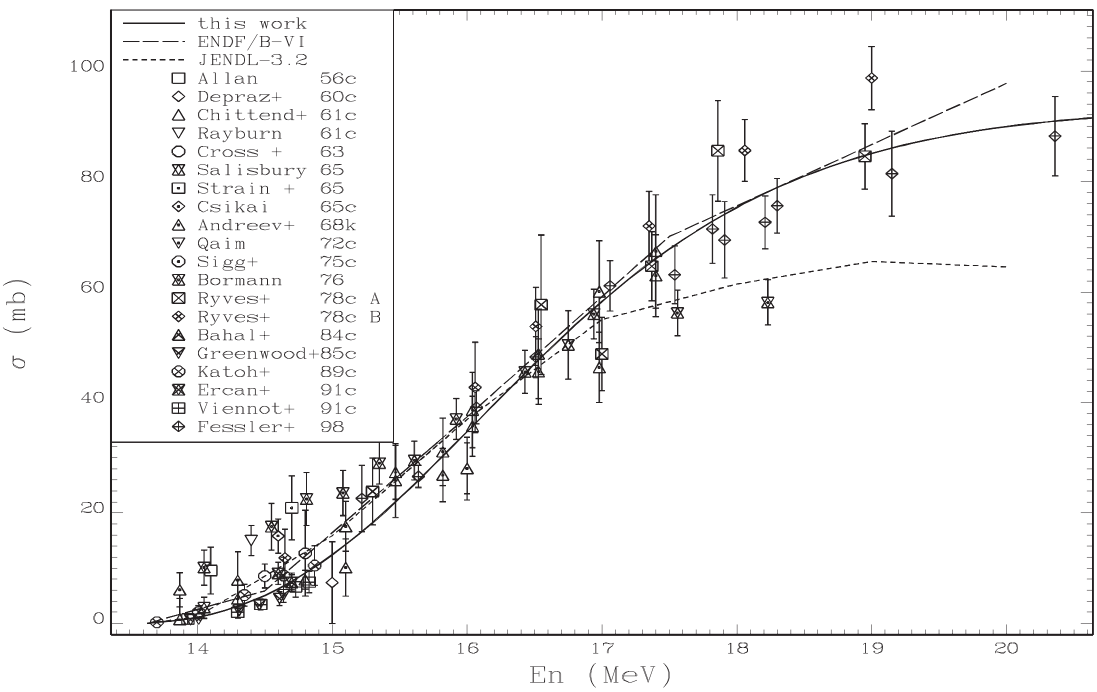

Fig. 6. The evaluated cross-section for the reaction $^{54}Fe(n,2n)^{53m+g}Fe$ in comparison with the experimental data and the JENDL-3 and ENDF/B-VI evaluations in the energy range from threshold to 20 MeV.

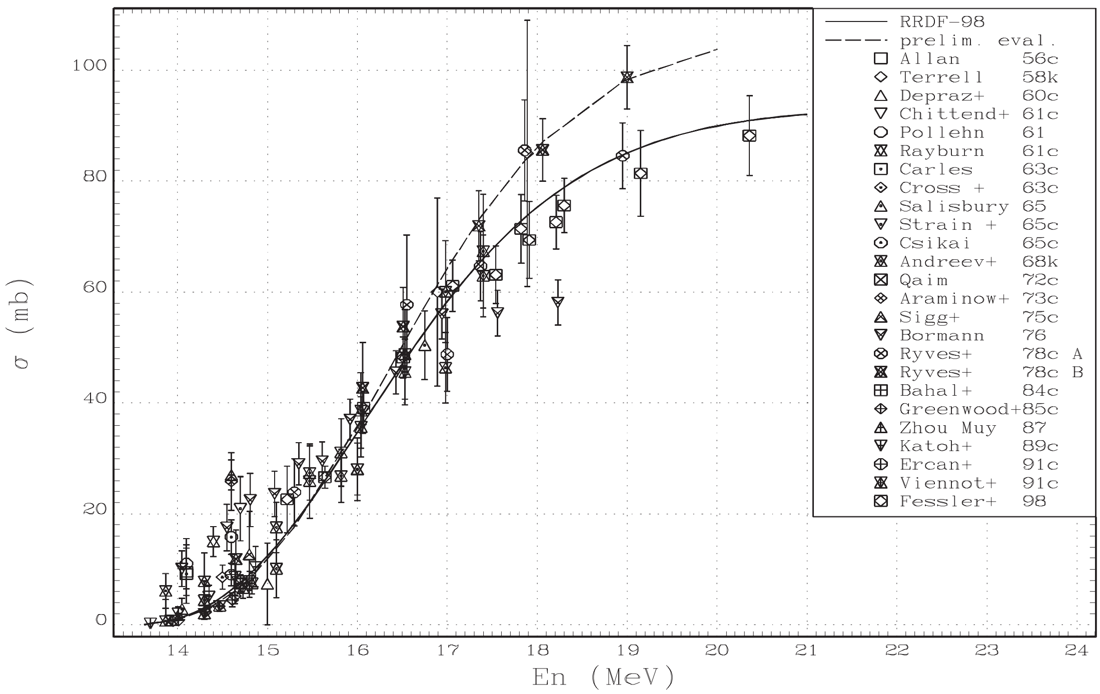

Fig. 7. The results of new evaluation in the comparison with data from preliminary FEI-93 evaluation [2].

THE CONSTANTS AND PARAMETERS OF NUCLEAR STRUCTURE AND NUCLEAR REACTIONS

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APPLICATION OF A MATHEMATICAL MODEL OF THE NUCLEUS TO PREDICT THE BINDING ENERGIES OF β-UNSTABLE HEAVY NUCLEI

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APPLICATION OF A MATHEMATICAL MODEL OF THE NUCLEUS TO PREDICT THE BINDING ENERGIES OF β-UNSTABLE HEAVY NUCLEI. A mathematical model for calculating heavy nucleus binding energies is developed. The model enables highly accurate calculation of the binding energies of all known nuclei (mean error 0.1 MeV) and is characterized by smooth extrapolation lines, ensuring very reliable extrapolation. A local increase in isobaric cross-section curvature is found, on the background of a steady decrease as the number of nucleons increases. This leads to a reduction in the binding energies of exotic nuclei compared to the results of other known estimation methods. The differences in binding energy reach 5-7 MeV close to the limits of the mathematical model.

Over the past few years interest in investigating the properties of β-unstable atomic nuclei has increased sharply. Facilities already doing work in this specialized area include Dubna, GANIL (Caen, France) and the University of Michigan, USA. New radioactive ion accelerators are also being built. The compilation of research programmes and their successful implementation require accurate and reliable prediction of β-unstable nucleus binding energies.

The existing mass formulas are unfortunately far from perfect, especially for nuclei that are removed from the stability band. The mathematical model (m. model) of the nucleus used in this paper, establishing formal links between the measured nucleon binding energies to the nuclei, has proved suitable for extrapolation and has a higher degree of accuracy. First done for medium and medium-heavy nuclei [1.2], this paper extends the method to heavy nuclei.

In the m. model of the nucleus, the space of the variables ZON determining the proton-neutron composition of the nucleus is divided by the (sub)magic numbers of the protons and neutrons N_l into rectangular fields. The part of the nuclei for which $Z_k \le Z \le Z_{k+1}$ and $N_l \le N \le N_{l+1}$ is called the inter(sub)magic field $\{\kappa, 1\}$. In each such field, the binding energies of the protons $p(Z,N)$ and of the neutrons $n(Z,N)$ to nuclei of the same type of parity i,j ($i = (-1)^Z$, $j = (-1)^N$) are linear functions of the nuclear variables Z and N:

$$p_{ij}(Z,N) = p_{ij}^0 - \tfrac{1}{2}\alpha_{ij}(z - z_k) + \tfrac{1}{2}\gamma_{ij}^N(N - N_l), \qquad (1)$$

$$n_{ij}(Z,N) = n_{ij}^0 + \tfrac{1}{2}\gamma_{ij}^Z(z - z_k) - \tfrac{1}{2}\beta_{ij}(N - N_l). \qquad (2)$$

Here p_{ij}^0 and n_{ij}^0 are the values of the binding energies of the nucleons to the fictitious nucleus (Z_k,N_l) and $\alpha_{ij}, \gamma_{ij}^Z, \gamma_{ij}^N, \beta_{ij}$ are parameters on which the conditions $\alpha_{i+} = \alpha_{i-} = \alpha_i$, $\beta_{+j} = \beta_{-j} = \beta_j$, $\gamma_{-j}^Z = \gamma_{+j}^Z = \gamma_j^Z$, $\gamma_{i-}^N = \gamma_{i+}^N = \gamma_i^N$ are imposed and, as a result of the continuity of the nuclear energy surface (NES) in the field $\{k, 1\}$, the following equation must be satisfied.

$$\gamma_+^N + \gamma_-^N = \gamma_+^Z + \gamma_-^Z$$

The complete system of parameters and values of the nucleon binding energies in nuclei which are required to reproduce the heavy nucleus m. model is shown schematically in Table 1. In this Table, each rectangle represents an inter(sub)magic field $\{\kappa, 1\}$. The order of the numbers beneath the field boundary lines is: the first denote the proton and neutron binding energies to the nucleus (Z_k,N_l), and the second the proton binding energies to the nucleus (Z_{k+1},N_l) and the neutron binding energy to the nucleus (Z_k,N_{l+1}). The single number above the boundary line Z_k denotes the proton binding energy to the nucleus (Z_k,N_{l+1}). The numbers inside the rectangles denote the values of the parameter $\gamma = \gamma_+^Z + \gamma_-^Z$ and the external a_i and b_j parameters are written above the arcs along the outer boundary of the heavy nucleus m. model. The experimental data for constructing the heavy nucleus m. model are taken from [3].

Table 1

Examples of complete parameter systems of the heavy nucleus mathematical model

N↑							
		0,85 0,72	0,83 0,89	0,85 1,00	0,92 0,98	0,87 0,85	
156	0,31 0,17	0,82	0,84	0,78	5,48 6,90 4,69 5,80 0,79 5,10	3,79 4,86 5,53 6,54 0,97 3,03	6,35 7,66
152	0,34 0,23	1,03	1,05	6,42 8,15 4, 56 5,50 0,96 5,99	4,82 5,9 8 5,52 6,46 0,98 4,25	2,95 3,7 6 6,48 7,46 1,00 2,05	7,46 8,48
148	0,38 0,22	0,97	0,98	5,68 6,97 5,12 6,15 0,91 5,21	3,88 4,9 6 6,10 6,99 0,93 3,24	1,93 2,78 6,90 8,05 0,93 1,51	7,82 8,99
144	0,33 0,26	1,00	6,50 8,12 4,58 5,92 1,02 6,03	4,88 5,95 5,68 6,86 0,93 4,33	2,94 4,0 4 6,70 7,70 0,93 2,30	1,21 1,64 7,76 8,50 0,95 0,20	8,70 9,46
140	0,28 0,24	1,18	5,54 7,04 4,87 6,48 1,18 5,00	3,92 5,05 6,11 7,60 1,08 3,31	1,88 3,24 7,13 8,74 1,12	0,11 0,84	
136	0,20 0,28	6,99 8,32 2,84 5,34 1,30 6,24	4,50 5,72 5,30 6,78 1,28 3,63	2,84 3,97 6,54 8,10 1,20 1,82	7.84 9.20 1,21		
132	0,19 0,09	5,49 7,22 3,98 5,24 1,28 4,34	3,00 4,66 5,66 7,40 1,10 1,72	1,60 2,81 6,68 8,58 1,05			
126	***82***	3,84 5,03 ***88***	1,32 3,04 ***92***	***96***	***100***	***104***	***Z***→

In order to extrapolate the binding energies in a field where the nuclei have not been analyzed experimentally, it is natural to use the m. model of even-even (e-e) nuclei and make parity corrections so that the number of m. model parameters can be reduced threefold. In the m. model of e-e nuclei, p^0, the binding energy of two protons to the nucleus (Z_k,N_l), is used instead of p_j^0 n^0, the binding energy of two neutrons to the nucleus (Z_k,N_l), is used instead of n_j^0, and $\alpha = \alpha_+ + \alpha_-$, $\beta = \beta_+ + \beta_-$ and $\gamma = \gamma_+^Z = \gamma_-^Z$ are used instead of $\alpha_i, \beta_j, \gamma_j^Z$. The total binding energies of e-e nuclei in the field $\{\kappa, l\}$ are expressed with a high degree of accuracy by the quadratic function from Z and N:

$$B(Z,N) = \frac{2p^0+\alpha}{4}(Z-Z_k) + \frac{2n^0+\beta}{4}(N-N_l) - \frac{\alpha}{8}(Z-Z_k)^2 - \frac{\beta}{8}(N-N_l)^2 + \\ + \frac{\gamma}{4}(Z-Z_k)(N-N_l) + B(Z_k,N_l) \tag{3}$$

for $Z_k \le Z \le Z_{k+1}$ and $N_l \le N \le N_{l+1}$.

Each paraboloid (3) is defined by the boundary values $B(Z_k,N_l)$, p^0, n^0 and the three universal parameters α, β and γ, two of which, α and β, are known and γ is not known. The geometrical properties of the paraboloids forming the entire m. model of the nuclear energy surface (NES) are predetermined to a significant extent by stable nuclei. This enables the law for variation in any geometrical parameter to be calculated and γ to be found.

The new parameter system reflecting the geometrical representations of the paraboloid properties may be given by the:

1) isobar curvature parameter $K = \dfrac{\alpha+\beta+2\gamma}{32}$;

2) orientation parameter of the isobaric maximum line projection to the plane *ZON*

$$P = \frac{ß+\gamma}{\alpha+\beta+2\gamma};$$

3) parameter characterizing the type of paraboloid $T = \dfrac{\sqrt{\alpha\beta}}{\gamma}$.

The most predictable of these is *P*.

In heavy nuclei, there are not enough experimental data to ensure reliable statistics for constructing extrapolation lines *P*(*Z*,*N*). The m. model parameters were therefore adjusted to eliminate oscillation in the calculated *P*({*k*,*l*}) in relation to the family of extrapolation lines *P*(*Z*,*N*). In the new variant of the m. model, the calculated nuclear binding energy values do not deviate from the experimental values in the interpolation fields by more than 0.1 MeV in 91% of cases, and the points P({*k*,*l*}) lay on the family of straight parallel lines (Fig. 1). As a result of the procedure described for extrapolating the m. model parameters in a field where the nuclei have not been analyzed experimentally, denoted in Fig. 1 by the dotted lines, tables of the atomic nuclear binding energies were drawn up (Tables 2-4). The upper numbers in the

Tables are the calculated nuclear binding energies and the lower numbers are the experimental values with the experimental error, rounded to the nearest hundredth MeV, included in brackets. Some of the data from Table 1 is contained in Fig. 2. in graphical form.

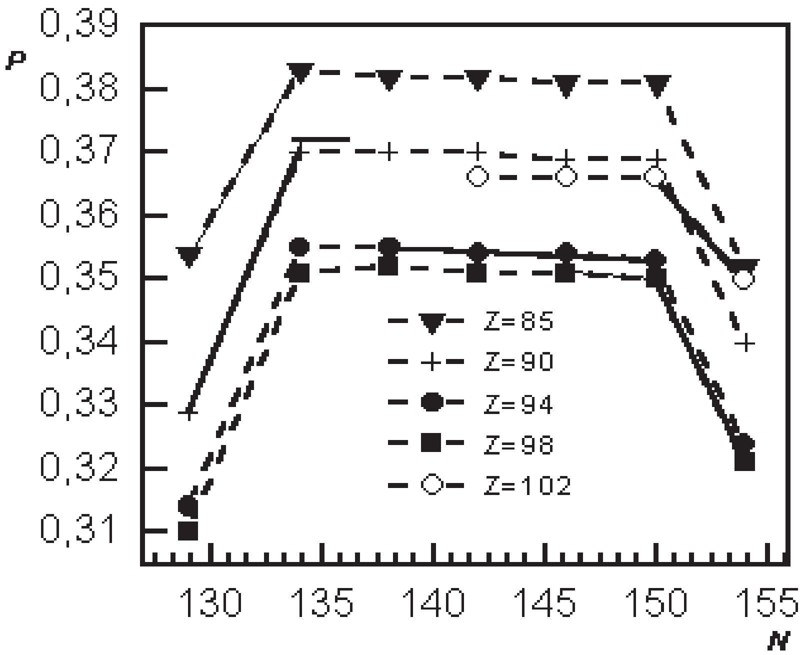

Fig. 1 Extrapolation line family chart for orientation parameter *P(Z/N)*

Table 2

Nuclear binding energies, MeV

								N ↑
1739,08		1764,39		1788,13		1810,30		***156***
1734,77		1759,26		1782,18		1803,53		*154*
1729,98		1753,65		1775,75		1796,28		***152***
1725,68		1748,32		1769,39		1788,89		*150*
1720,81		1742,42		1762,46		1780,93		***148***
1715,01		1735,65		1754,72		1772,22		*146*
1708,61		1728,28		1716,38		1762,01	1769,41	***144***
						1757,25	1763,76 1763,77(10)	*143*
1701,70		1720,37		1737,47		1753,00 1753,01(3)	1759,02 1758,94(10)	*142*
						1747,08 1747,05(6)	1753,11 1753,22(10)	*141*
1694,20		1711,87		1727,97		1742,50 1742,48(1)	1748,04 1748,03(5)	***140***
						1736,26 1736,17(1)	1741,78 1741,74(1)	*139*
1686,91		1703,40		1718,32		1731,67 1731,61(1)	1736,69 1736,71(1)	*138*
						1725,19 1725,21(1)	1730,19 1730,19(1)	*137*
1679,10	1686,09	1694,41	1700,55	1708,15 1708,19(1)	1713,44 1713,46(1)	1720,32 1720,31(1)	1724,82 1724,79(1)	***136***
1674,04	1681,03	1688,87	1695,01	1702,13	1707,42 1707,43(2)	1713,82 1713,83(1)	1718,20 1718,13(1)	*135*
1671,40	1677,64	16,85,41 1 685,48(1)	1690,80 1690,60(8)	1697,85 1697,80(1)	1702,39 1702,43(1)	1708,72 1708,68(1)	1712,47 1712,50(1)	*134*
1666,06	1672,30	1679,59	1684,93 1684,96(1)	1691,55 1691,51	1696,09 1696,16(1)	1701,94 1701,96(1)	1705,57 1705,61(2)	*133*
1663,2 2 1663,30(1)	1668,71 1668,69(10)	1670,93 1675,91(1)	1680,57 1680,59(1)	1687,07 1687,06(1)	1690,86 1690,95(1)	1696,64 1696,58(1)	1699,64 1699,62(5)	***132***
1658,16	1663,60 1663,54(1)	1670,15 1670,16(1)	1674,74 1674,66(1)	1680,57 1680,55(1)	1684,31 1684,44(1)	1689,42 1689,40(1)	1692,26 1692,31(5)	*131*
1654,56 1654,52(1)	1659,50 1659,50(1)	1665,99 1666,03(1)	1670,08 1670,10(1)	1675,85 1675,88(1)	1679,09 1679,11(1)	1684,14 1684,07(1)	1686,58 1686,43(5)	*130*
1649,41 1649,39(1)	1654,30 1654,32(1)	1660,12 1660,14(1)	1664,16 1664,16(1)	1669,26 1669,24(1)	1672,45 1672,37(1)	1676,83 1676,76(1)	1679,11 1679,09(5)	*129*
1645,62 1645,57(1)	1650,01 1649,98(1)	1655,77 1655,79(1)	1659,31 1659,28(1)	1664,35 1664,32(1)	1667,04 1666,97(1)	1671,36 1671,28(1)	1673,24 1673,15(1)	*128*
1640,33 1640,38(1)	1644,72 1644,85(1)	1649,81 1649,78(1)	1653,30 1653,25(1)	1657,67 1657,62(1)	1660,31 1660,17(1)	1663,96 1663,97(1)	1665,68 1665,66(2)	*127*
1636,40 1636,44	1640,24 1640,02(1)	1645,27 1645,23(1)	1648,26 1648,21(1)	1652,57 1652,51(1)	1654,71 1654,69(1)	1658,30 1658,33(1)	1659,62 1659,70(1)	***126***
82	*83*	*84*	*85*	*86*	*87*	***88***	*89*	***Z***→

Table 3

Nuclear binding energies, MeV

							N ↑
1830,66		1849,30		1865,43		1879,71	***156***
						1874,08 1874,15(1)	*155*
1823,05		1840,85		1856,20		1869,70 1869,74)	*154*
						1863,90 1863,91(1)	*153*
1814,96		1831,96	1838,34	1846,49 1846,62(2)	1852,06	1859,21 1859,20	***152***
		1826,65	1833,01	1840,74 1840,84(2)	1846,25 1846,23(2)	1352,98 1852,98(1)	*151*
1806,52		1822,43	1828,48 1828,65(1)	1836,04 1836,06(1)	1841,24 1841,26(1)	1847,80 1847,83(1)	*150*
		1816,93	1822,92 1823,09(20)	1830,06 1830,04	1835,20 1835,20(1)	1841,34 1841,37(1)	*149*
1797,51		1812,37 1812,43(1)	1818,05 1818,17(7)	1825,02 1825,01(1)	1829,85 1829,84(1)	1835,82 1835,85(1)	***148***
	1799,02(6)	1806,44 1806,50(1)	1812.05 1812,04(2)	1818,67 1818,70(1)	1823,43 1623,47(1)	1829,05 1829,05(1)	*147*
1787.82	1794,07(10)	1801,70 1801,69(1)	1806.98 1806,98(1)	1813.44 1813,46(1)	1817.87 1817,94(1)	1823,33 1823,36(1)	*146*
1782,10(5)	1788,30(20)	1795,55 1795,54(1)	1800,76 1800,77(1)	1806,87 1806,92(1)	1811,23 1811,29(2)	1816,34 1816,39(1)	*145*
1777,53 1777,67(1)	1783,20 1783,24(5)	1790,43 1790,42(1)	1795,31 1795,28(1)	1801,26 1801,26(1)	1805,29 1805,34(1)	1810,24 1810: 30(1)	***144***
1771,40 1771,48(1)	1777,08 1777,16(1)	1783,83 1783,87(1)	1788,64 1788,69(5)	1794,24 1794,28(1)	1798,20 1798,24(5)	1802,80	*143*
1766,60 1766,69(1)	1771,79 1771,94(1)	1778,48 1778,57(1)	1782,88 1782,97(1)	1788,38 1788,40(1)	1791,93	1796,43 1796,48(4)	*142*
1760,21 1760,25(1)	1765,41 1765,41(1)	1771,62 1771,73(1)	1775,95 1775,98(1)	1781,10	1784,58	1788,7 3	*141*
1755,08 1755,14(1)	1759,79 1759,86(1)	1765,94 1765: 97(1)	1 769,86	1774,91 1774,81(1)	1777,98	1782,03	***140***
1748,28 1748,34(1)	1752,97 1753,04(1)	1758,58 1758,72(5)	1762,43	1766,98 1767,03(5)	1769,98	1773,53	*139*
1743,07 1743,09(1)	1747,26 1747,24(1)	1752,75 1752,82(1)	1756,13 1756,10(5)	1760,64 1760,65(2)	1763,17	1766,68	*138*
1736,03 1735,98(1)	1740,20 1740,19(1)	1745,15 1745,15(1)	1748,46 1748,43(6)	1752,47	1754,93	1757,94	*137*
1730,54 1730,52(1)	1734,21 1734,18(1)	1739,04 1739,07(2)	1741,88 1741,81(9)	1745,85 1745,94(3)	1747,84	1750,81	***136***
1723,38 1723,33(1)	1726,93 1726,91(2)	1731,22 1731,21(2)	1733,66	1737,48	1739,07	1741,89	*135*
1717,66 1717,58(1)	1720,58 1720,53(7)	1724,88 1724,82(2)	1727,10	1730,49	1731,86	1734,25	*134*
1710,22 1710,13(1)	1713,02 1712,92(5)	1716,78 1716,70(5)	1718,60	1721,84	1722,81	1725,05	*133*
1704,30 1704,23(1)	1706,47 1706,39(7)	1710,24 1710,30(3)	1711,84	1714,65	1715,40	1717,21	***132***
1696,49 1696,42(1)	1698,50	1701,84					*131*
1690,70 1690,62(2)	1692,31 1692,26(5)	1695,54		1698,90		1700,41	*130*
1682,80 1682,75(5)	1684,25 1684,13(6)	1687,05					*129*
1676,82 1676,78(2)	1677,87 1677,91(7)	1680,56		1682,87		1683,33	*128*
1668,53 1668,90(3)	1669,72 1669,71(7)	1671,98					*127*
1662,66 1662,70(2)	1663,15 1663,25(7)	1665,30		1666,56		1665,97	***126***
90	*91*	***92***	*93*	*94*	*95*	***96***	***Z***→

Dripline

Table 4
Nuclear binding energies, MeV

								N↑
1885,19	1892,09 1892,11(1)	1896,65 1896,65(1)	1902,57 1902,54(1)	1906,36 1906,32(1)	1911,22	1914,14	1918,15	***156***
1879,51	1886,09 1886,08(1)	1890,60 1890,68(1)	1896,20 1896,16(1)	1899,65 1899,63(5)	1904,29 1904,31(3)	1906,87	1910,66	*155*
1874,85 1874,79(1)	1881,29 1881,28(1)	1885,52 1885,58(1)	1890,98 1890,98(1)	1894,35 1894,33(1)	1898,66 1898,64(1)	1901,16	1904,62	*154*
1869,00 1869,00(1)	1875,12 1875,10(1)	1879,30 1879,23(5)	1884,44 1884,47(1)	1887,47	1891,56 1891,54(1)	1893,72	1896,96	*153*
1864,03 1864,03(1)	1870,01 1870,00(1)	1873,91 1873,94(1)	1878,91 1878,93(1)	1881,86	1885,62 1885,60(2)	1887,70	1890,61 1890,65(3)	***152***
1857,70 1857,70(2)	1863,28 1863,37(1)	1867,08	1871,68 1871,69(1)	1874,24	1877,88	1879,57	1882,36	*151*
1852,15 1852,25(1)	1857,62 1857,78(1)	1861,05	1865,54 1865,53(1)	1867,98	1871,25 1871,30(1)	1872,82	1875,24	*150*
1845,59 1845,70(6)	1850,66 1850,82(1)	1353,99 1853,94(6)	1858,08	1860,13	1863,28	1864,46	1866,76	*149*
1839,70 1839,78(1)	1844,66 1844,79(1)	1847,62	1851,60 1851,56(1)	1853,53	1856,31	1857,37	1859,30	***148***
1832,76 1832,81(5)	1837,36 1837,43(1)	1840,15	1843,77					*147*
1826,74 1826,76(1)	1831,24 1831,26(1)	1833,73	1837,25 1837,19(4)		1841,03		1843,09	*146*
1819,58	1823,72	1826,04	1829,20					*145*
1813,18	1817,22	1819,24	1822,30		1825,15		1826,28	***144***
1805,63	1809,38	1811,29	1814,06					*143*
1798,84	1802,48	1803,97	1806,63		1808,49		1808,67	*142*
1791,03	1794,38	1795,76	1798,13					*141*
1783,91	1787,15	1788,11	1790,37		1791,24		1790,47	***140***
	1770,68		1772,78					*138*
	1753,69		1754,67					***136***
	1735,92		1735,69					*134*
	1717,67		1716,23					***132***
97	*98*	*99*	***100***	*101*	*102*	*103*	***104***	*Z*→

Dripline

The heavy nucleus m. model is limited from above by the N number $N_l = 156$ which has the characteristic of a magic number sign with respect to the sharp reduction in the value P, which is characteristic for magic numbers. The m. model is limited by the number $Z_k = 104$ because of the lack of reliable experimental data for determining the a-parameters. A proton stability line is obtained for proton-rich nuclei.

The heavy nucleus m. model has special features compared with lighter nuclei: 1) local increase in the α- and β-parameters against the background of their systematic decrease for the whole of the NES as Z and N increase; and 2) on the whole, the NES has asymmetrical isobaric cross-sections. (Asymmetry occurs because the different rate of change in the proton and neutron energy levels in the isobaric cross-section, due to the inequality ($\frac{\partial \alpha}{\partial Z} > \frac{\partial \beta}{\partial N}$) caused by the Coulomb energy of the nuclei). This trend is also observed in heavy nuclei owing to the inequality $\frac{\partial}{\partial I}\left(\sqrt{\alpha}+\sqrt{\beta}\right) > 0$ for $82 \leq Z \leq 100$ (where $I = N\text{-}Z$) [2]. Unlike medium nuclei, the discrepancy between the β-stability lines and the dividing line of paraboloids of different types makes it difficult to prove directly that the heavy nucleus isobaric cross-sections are asymmetrical.

The binding energies obtained in the m. model for nuclei removed from the band of known nuclei were lower than in [5, 6]. Around the proton stability line, the nuclear binding energies calculated in this paper are 1-2 MeV lower than in [5, 6], and in neutron-rich nucleus fields their binding energies are systematically lower compared with similar energies in [5] for N>146 and in [6] for N>140, with the difference reaching 5.7 MeV in ^{238}Pb. Similar trends in the binding energy in this paper are observed in relation to [4], although they are much weakened by the 1-3 MeV reduction in the binding energy of stable nuclei in [4]. The reduction in the binding energies of exotic nuclei corresponds to the 10-15% increase in curvature of the isobaric cross-sections in the m. model which is due to local increase in the α and β parameters in heavy nuclei. In accordance with the increase in curvature of the isobaric cross-sections, the nucleon stability boundaries are located closer to the β-stability line.

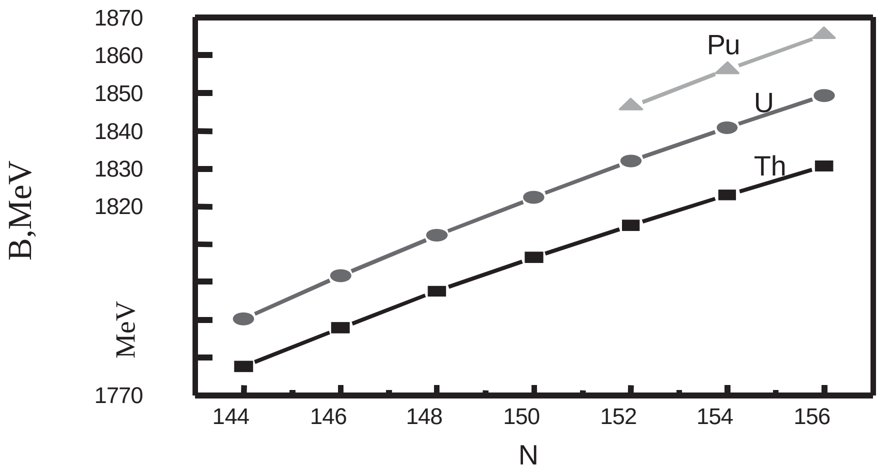

Fig. 2. Binding energies of the neutron-rich nuclei Th, U, P_u

In conclusion, I wish to thank Mr F.E. Churkeev for his valuable suggestions and comments during preparation of the article for publication.

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NUCLEAR REACTOR DATA

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THE FOND-2.2 EVALUATED NEUTRON DATA LIBRARY

(Russian library of evaluated neutron data files for generating sets of constants in the ABBN constants system)

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THE FOND-2.2 EVALUATED NEUTRON DATA LIBRARY. A short description is given of the Russian evaluated neutron data library FOND-2.2. The main purpose of FOND-2.2 is to provide sets of constants for the ABBN constants system. A history of its compilation and the sources of the neutron data are given. The contents of FOND-2.2 are presented with brief comments.

The FOND library contains evaluated neutron data files for a large number of nuclides. The library's main purpose is to provide the sets of neutron data needed to create libraries of group constants for the ABBN constants system [1]. The library was created in the ABBN constants laboratory of the A.I. Leipunsky Institute for Physics and Power Engineering. The new version of the ABBN-93 group constants system, which was created using the FOND library, is used widely in a number of applications:

- calculations for nuclear physics facilities with different spectral characteristics;
- radiation shielding calculations;
- nuclear safety problems, etc.

History

1970 - During the development of the SOKRATOR system [2], work began on the compilation of library evaluated neutron data in machine format for the main reactor material ^{238}U and a number of other nuclides.

1984 - The compilation of the first version of the FOND library, which formed part of the SOKRATOR system for providing group constants for nuclear reactor calculations, was completed. Initially, the library contained evaluated neutron data for the 67 nuclides of most importance for fast reactor calculations [3]. The library was the principal (but not the only) neutron data source used in creating the ABBN-78 group constants library [4].

1988 - During the development of the new ABBN-90 constants system [5], the first fairly complete version of the evaluated neutron data library FOND-1 was compiled; it contained neutron data for 121 nuclides [6]. The library included neutron data for the main fuel materials, shielding materials, moderators, fission products and minor actinides.

1993 - As part of the process of verifying the ABBN-90 constants system by comparing the results of calculations with the results obtained in critical assemblies and the BN-350 power reactor, a review was carried out of neutron data in the FOND library for a number of basic fuel nuclides and structural materials. In particular, the ENDF/B-VI (Rev.2) evaluation was adopted for ^{235}U and the JENDL-3 library evaluation - modified in the low-energy region - for ^{239}Pu, the neutron data for Ni, Na, Pb, O, N and C were reviewed and evaluations for new materials were added to the library which then contained evaluated neutron data for 268 nuclides [7]. This version was named FOND-2.

1996 - Version FOND-2.1a was produced. It included evaluated neutron data for 52 minor actinides; these data were selected while the actinide library was being created in the ABBN-93 constants system [8] following analysis of a series of international tests [9] and taking into account available experimental data [10].

1998 - Version FOND-2.1b was produced. It included evaluated neutron data for 172 fission product nuclides selected while the fission product library was being created in the ABBN-93.1 constants system and verified in the SWG-17 international test [11].

1999 - Version FOND-2.2 was produced and finalized which forms the basis of the modern ABBN-93.2 group constants system and contains the earlier data for the main reactor materials and improved data for fission products and minor actinides. The FOND-2.2 library contains evaluated neutron data for 452 nuclides.

Neutron data sources

Traditionally, the main source of neutron data for the FOND library has been the BROND national evaluated neutron data library [6, 7] compiled in the Russian Nuclear Data Centre with the participation of specialists from the ABBN constants laboratory. In the first phase, evaluated neutron data from the foreign ENDF/B, JENDL, JEF and EAF libraries, and later from the Russian ADL library, were used when gaps existed in the BROND library.

At that stage, nuclear data were selected from foreign libraries by means of a comparative analysis of their simplest integral characteristics, such as:

- reaction cross-sections at the thermal point (E = 0.0253 eV);
- resonance integrals;
- cross-sections averaged over known standard spectra, etc.

In the second phase, when selecting neutron data for some of the main reactor nuclides, the results of calculation-based studies of experiments performed in a number of critical assemblies and operational reactors were taken into account. The calculations employed different sets of group constants obtained using several of the neutron data files being studied.

For later versions of the FOND library, emphasis was placed on the up-to-dateness and completeness of the experimental data used for the evaluation when selecting evaluated neutron data, and compliance with the requirement for congruence with the results of integral experiments.

Data presentation format

As a component of the SOKRATOR system, the FOND library was at first produced in the specially developed SOKRATOR format [12], which is similar to the UKNDL format [13]. Subsequently, as the FOND library developed, the following ENDF formats were used to store the evaluated neutron data [14]:

ENDF-IV format (for the first version FOND-1);
ENDF-V format (for FOND-2);
ENDF-VI format (for FOND-2.1 and FOND-2.2).

The only difference from the ENDF-IV format is the rule for generation of the material identifier MAT.

MAT is a four-digit identifier of the type ZZAA. As is the case in the ENDF format, the highest-order digits ZZ contain the atomic number **Z**. The two lowest-order digits AA, **as a rule**, contain the two lowest-order digits of the mass number of the nuclide **A**. For natural mixtures, the latter are zeros. For example: for ^{235}U, MAT = 9235; for ^{3}He, MAT = 0203; and for ^{nat}Pb, MAT = 8200.

The exceptions are as follows:

a) for ^{100}Mo, MAT = 4210; for ^{100}Ru, MAT = 4410; for ^{200}Hg, MAT = 8020, etc.;

b) when generating the last two digits of the MAT identifier for first isomers, the mass number is decreased by 30, and for second isomers it is decreased by 40. Thus: for ^{242m}Am, MAT = 9512; and for ^{178n}Hf, MAT = 7238, etc.

Use of computer programs

The neutron data from the FOND library can be processed by any computer program which processes data in ENDF format.

The following programs are usually used:

- The Utility Codes program package [15] is traditionally used to verify the consistency of the data from the point of view of the format and the physics involved. Different versions of this program package have been used as the FOND library has developed;

- The GRUKON application package [16] and the NJOY94 neutron data processing system [17] have traditionally been used to process the evaluated neutron data into the required sets of group constants;

- The ENDF/B Pre-Processing Codes program package may also be used to process neutron data [18], but the authors have only limited practical experience of its use.

Structure of the library

A diagram of the library's structure is shown in Fig. 1. The library consists of four sections: neutron data for light elements, structural materials and intermediate nuclei, and data for actinides.

The neutron data can be classified into data for main and secondary materials. For main materials, all the neutron data required for neutron field calculations are given. These are the general purpose (GP) files. For secondary materials, often only the data on neutron reactions which bring about a change in the material's composition are given. These are the activation and transmutation data (ACT, MA or FP). The latter type are used to calculate activation of reactor configurations, the nuclide composition of nuclear fuel after its removal from the reactor, cooling, chemical processing, recycling, disposal, etc.

Composition of the neutron data files

The files contain different types of neutron data. The full range is not given for all nuclides. Generally, the files contain the following:

data on neutron transport	(MF = 2, 3, 4, 5);
energy-angle data	(MF = 6);
decay data	(MF = 8, 9, 10);
data on photon production	(MF = 12, 13, 14, 15);
data on errors and their covariances	(MF = 32, 33 etc.).

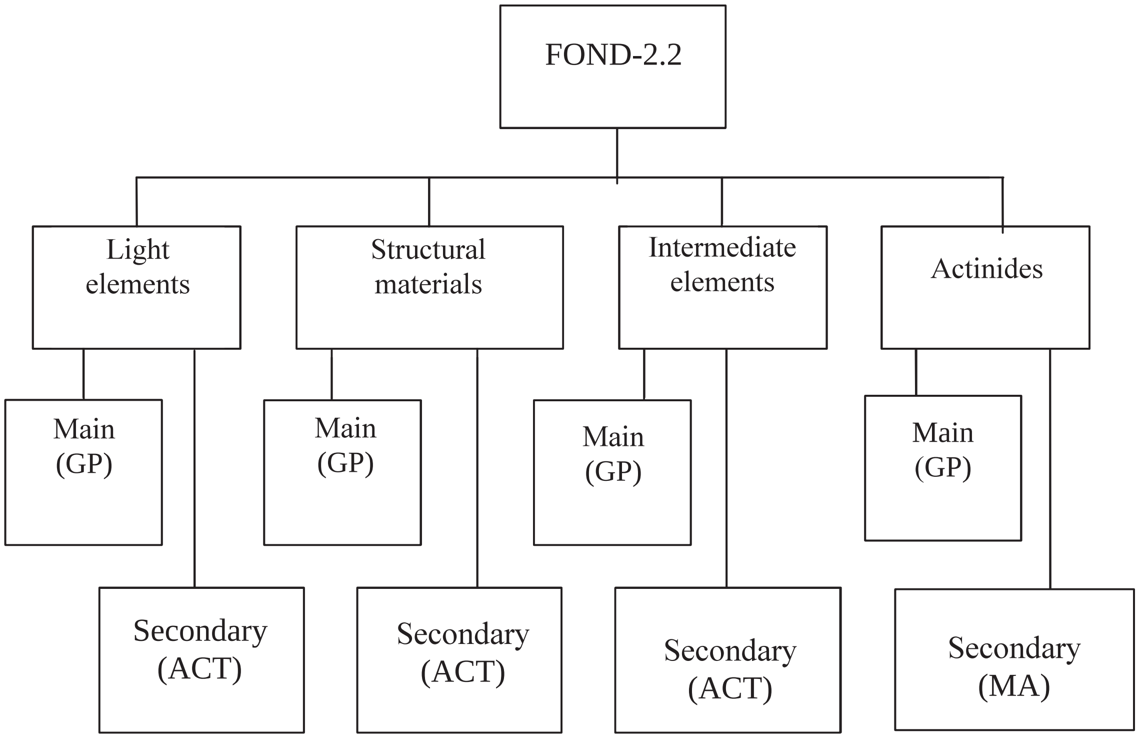

Fig. 1. Structure of the FOND-2.2 library

It should be noted that the decay data file (MF = 8) is always minimal in size, i.e. it gives the radionuclides which form as a result of neutron reactions but not the radiation characteristics of those radionuclides. MF = 8 files are given only for materials for which there are also MF = 9 or MF = 10 files.

Contents of the library

A list of the nuclides for which the FOND-2.2 library contains data is given in the table. Some explanations of the table are given below.

NUCLIDE - The symbols of natural nuclides are shown in bold.

For radioactive isotopes, the half-life is given and the following symbols are used:

y - years; d - days; h - hours; m - minutes, in accordance with the nuclide chart in [19].

SOURCE - Original evaluation from which neutron data were taken for FOND-2.2;

EVAL/REV - Date of neutron data evaluation/date when data revised;

STATUS - Recommended area of application:

GP general-purpose neutron data;
ACT neutron data for use in calculating activation;
FP neutron data for fission products;
MA neutron data for minor actinides.

COMPOSITION OF DATA:

NT main data required for neutron transport calculations;
DDD energy-angle data;
DD decay data;
GAM data on photon production;
COV data on errors in cross-sections or the parameters describing them, and on the covariances of these errors.

COMMENTS

Reference is frequently made to the existence (availability) of a later evaluation of the data for a given nuclide. This does not mean that it is advisable to adopt that evaluation; it merely indicates the desirability of comparative study of existing evaluated data, and of taking decisions on the results of such a comparison.

Contents of the FOND-2.2 library

LIGHT ELEMENTS

No.	Nuclide	Source	EVAL/REV	Status	Composition of data and brief comments
1	**1-H-1**	ENDF/B-6 Rev. 1	Oct89/Jul91	GP	NT, GAM **Total cross-section below 20 MeV recommended as a standard. Rev. 3 (1999) available with slight corrections.**
2	**1-H-2**	ENDF/B-6 Rev. 2	Nov67/Nov96	GP	NT, DD, GAM Rev. 3 (1999) available with slight corrections.
3	*1-H-3 12.323 y*	BROND-2	Dec 88/	GP	NT, DDD
4	**2-He-nat**	FOND-2	Jan76/Jan92	GP	NT
5	**2-He-3**	BROND-2	Dec 88/	GP	NT, DDD, DD **Cross-section for (n,p) reaction up to 50 keV recommended as a standard.** Corresponds to the data in the ENDF/B-6 library (Rev. 1).
6	**2-He-4**	BROND-2	Jan76/Jan92	GP	NT
7	**3-Li-6**	BROND-2	Jan76/Jan92	GP	NT, DDD, COV **Cross-section for (n, t) reaction up to 1 MeV recommended as a standard.** Corresponds to the data in the ENDF/B-6 library (Rev. 1).

No.	Nuclide	Source	EVAL/REV	Status	Composition of data and brief comments
8	**3-Li-7**	BROND-2	May84/Feb92	GP ACT	NT, DD, GAM
9	**4-Be-9**	ENDF/B-6	Jan86/	GP ACT	NT, DD, GAM
10	*4-Be-10* *1.6×10^{6} y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
11	**5-B-10**	ENDF/B-6 Rev. 1	Oct89/Jul91	GP ACT	NT, GAM **Cross-sections for (n,α) and (n,α1) reactions up to 100 keV recommended as standards.**
12	**5-B-11**	ENDF/B-6	May89/	GP ACT	NT, DD, GAM
13	**6-C**	ENDF/B-6 Rev. 1	Aug89/Jul91	GP	NT, GAM, COV **Elastic scattering cross-section up to 2 MeV, recommended as a standard.** Rev. 2 (1999) available with slight corrections.
14	**6-C-12**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
15	**6-C-13**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
16	*6-C-14* *5. 730 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
17	**7-N-14**	ENDF/B-6	May90/	GP	NT, GAM Rev. 3 (1999) available with re-evaluation in the resonance region.
18	**7-N-15**	ENDF/B-6	Sep83/	GP	NT, GAM
19	**8-O-16**	ENDF/B-6	Jun90/	GP	NT, GAM Rev. 1 (1999) available with slight changes below 20 MeV
20	**8-O-17**	ENDF/B-6	Jan78/	GP	NT, DD, GAM, COV
21	**8-O-18**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
22	**9-F-19**	ENDF/B-6	Jun90/	GP	NT, DD, GAM, COV
23	**10-Ne-20**	EAF-3	Jul92/	GP	Only sets of cross-sections given.
24	**10-Ne-21**	EAF-3	Jul92/	GP	Only sets of cross-sections given.
25	**10-Ne-22**	EAF-3	Jul92/	GP	Only sets of cross-sections given.
26	*11-Na-22* *2.603 y*	EAF-3	Jul92/	GP	Only sets of cross-sections given.
27	**11-Na-23**	ENDF/B-6 Rev. 1	Dec77/Jul91	GP ACT	NT, DD, GAM
28	**12-Mg**	JENDL-3.2	Mar87/Nov93	GP	NT, GAM
29	**12-Mg-24**	JENDL-3.2	Mar87/Apr93	GP	NT
30	**12-Mg-25**	JENDL-3.2	Mar87/	GP	NT
31	**12-Mg-26**	JENDL-3.2	Mar87/	GP	NT
32	*13-Al-26* *7.16×* *2.603 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
33	**13-Al-27**	ENDL-3	Mar88/	GP ACT	NT, GAM **Cross-section for (n,α) reaction recommended as a standard.** Corresponds to the data in the ENDF/B-6 library. JENDL-3.2 evaluation (1993) available.
34	**14-Si**	BROND-2	May85/	GP	NT, GAM
35	**14-Si-28**	ENDF/B-6	May96/	GP	NT, DD, GAM
36	**14-Si-29**	ENDF/B-6	May96/	GP	NT, DD, GAM
37	**14-Si-30**	ENDF/B-6	May96/	GP	NT, DD, GAM

No.	Nuclide	Source	EVAL/REV	Status	Composition of data and brief comments
38	*14-Si-31 2.62 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
39	*14-Si-32 172 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
40	**15-P-31**	BROND-2	May89/Dec90	GP ACT	NT, GAM Modification of ENDL-84 evaluation
41	*15-P-32 14.26 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
42	*15-P-33 25.34 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
43	**16-S**	ENDF/B-6	Apr79	GP	NT, GAM
44	**16-S-32**	JENDL-3.2	May87/Feb94	GP	NT
45	**16-S-33**	JENDL-3.2	May87/Feb94	GP	NT
46	**16-S-34**	JENDL-3.2	May87/Feb94	GP	NT
47	*16-S-35 87.5 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
48	**16-S-36**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
49	**17-Cl**	BROND-2	Feb90/	GP	NT, GAM Modification of ENDF/B-4 evaluation
50	**17-Cl-35**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
51	*17-Cl-36 $3.0{\times}10^5$ y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
52	**17-Cl-37**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
53	**18-Ar**	ENDL-84	Oct83/	GP	NT, GAM
54	**18-Ar-36**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
55	*18-Ar-37 35.0 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
56	**18-Ar-38**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
57	*18-Ar-39 269 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
58	**18-Ar-40**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
59	*18-Ar-41 1.83 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
60	*18-Ar-42 33 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
61	**19-K**	JENDL-3.0	May87/Jan93	GP	NT JENDL-3.2 (1994) evaluation available with major modification of cross sections.
62	**19-K-39**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
63	**19-K-40 $1.28{\times}10^9$ y**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
64	**19-K-41**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
65	*19-K-42 12.36 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
66	*19-K-43 22.2 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
67	**20-Ca**	FOND-2	Jan78/Dec90	GP	NT, GAM Modification of ENDL-78 evaluation.
68	**20-Ca-40**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
69	*20-Ca-41 $1.03{\times}10^5$ y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
70	**20-Ca-42**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.

No.	Nuclide	Source	EVAL/REV	Status	Composition of data and brief comments
71	**20-Ca-43**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
72	**20-Ca-44**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
73	*20-Ca-45 163 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
74	**20-Ca-46**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
75	*20-Ca-47 4.54 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
76	**20-Ca-48**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
77	*21-Sc-44m 2.44 d*	ADL-3	Jan94/	ACT	Only sets of cross-sections given.
78	**21-Sc-45**	JENDL-3.0	Feb82/Nov83	GP ACT	NT JENDL-3.2 (1993) evaluation available with major modification of cross-sections.
79	*21-Sc-46 83.82 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
80	*21-Sc-47 3.35 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
81	*21-Sc-48 43.67 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.

STRUCTURAL MATERIALS

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
1	**22-Ti**	FOND-2	Jun77/Dec90	GP	NT, GAM Modification of ENDL-78 evaluation.
2	*22-Ti-44* *47.3 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
3	*22-Ti-45* *3.08 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
4	*23-V-48* *15.97 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
5	*23-V-49* *330 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
6	**23-V-50** **1.4×10^{17} y**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
7	**23-V-51**	FOND-2	May77/Dec90	GP	NT, GAM Modification of ENDL-78 evaluation.
8	**24-Cr**	BROND-2	Apr84/Oct89	GP	NG, GAM
9	**24-Cr-50**	BROND-2	Apr85/Mar92	GP	NT, DD
10	*24-Cr-51* *27.70 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
11	**24-Cr-52**	BROND-2	Apr85/Mar92	GP	NT, DD
12	**24-Cr-53**	BROND-2	May88/Mar92	GP	NT, DD
13	**24-Cr-54**	BROND-2	May85/Mar92	GP	NT, DD
14	*25-Mn-52* *5.6 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
15	*25-Mn-53* *3.7×10^{6} y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
16	*25-Mn-54* *312.2 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
17	**25-Mn-55**	JENDL-3	Mar87/Mar88	GP ACT	NT, GAM, COV JENDL-3.2 (1993) evaluation available with modification of cross-sections.
18	**26-Fe**	BROND-2	Nov85/Mar90	GP	NT, DDD, GAM
19	**26-Fe-54**	BROND-2	Nov85/Nov90	GP	NT
20	*26-Fe-55* *2.73 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
21	**26-Fe-56**	BROND-2	Nov85/Nov90	GP	NT
22	**26-Fe-57**	BROND-2	Nov85/Oct89	GP	NT
23	**26-Fe-58**	BROND-2	Nov85/Nov90	GP	NT
24	*26-Fe-59* *44.503 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
25	*26-Fe-60* *1.5×10^{6} y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
26	*27-Co-56* *77.26 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
27	*27-Co-57* *271.79 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
28	*27-Co-58* *70.86 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
29	**27-Co-59**	FOND-2	Nov82/Jan84	GP	NT, modification of JENDL-2 evaluation. JENDL-3.2 evaluation (1994 mod.) available. **Cross-section for (n, 2n) reaction recommended as a standard**. Corresponds to the data in the ENDF/B-6 library.
30	*27-Co-60 5.272 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
31	*28-Ni-56 6.075 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
32	*28-Ni-57 36.0 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
33	**28-Ni-58**	ENDF/B-6 Rev. 1	Oct89/Jul91	GP, ACT	NT, DDD, GAM, COV Rev. 2 (1999) available.
34	*28-Ni-59* 7.5×10^4 *y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
35	**28-Ni-60**	ENDF/B-6 Rev. 1	Oct89/Jul91	GP, ACT	NT, DDD, GAM, COV Rev. 2 (1999) available.
36	**28-Ni-61**	ENDF/B-6	Feb89/	GP, ACT	NT, DDD, GAM, COV Rev. 3 (1999) available with slight changes below 20 MeV.
37	**28-Ni-62**	ENDF/B-6 Rev. 1	Oct89/Jul91	GP, ACT	NT, DDD, GAM, COV Rev. 4 (1999) available with slight changes below 20 MeV.
38	*28-Ni-63 100 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
39	**28-Ni-64**	ENDF/B-6 Rev. 1	Oct89/Jul91	GP, ACT	NT, DDD, GAM, COV Rev. 2 (1999) available.
40	*28-Ni-65 2.52 h*	ADL-3	Jan94/	ACT	Only sets of cross-sections given.
41	*28-Ni-66 54.6 h*	ADL-3	Jan94/	ACT	Only sets of cross-sections given.
42	**29-Cu-63**	ENDF/B-6	Nov89/	GP, ACT	NT, DDD, GAM, COV Rev. 3 (1999) available with slight changes below 20 MeV.
43	**29-Cu-65**	ENDF/B-6	Nov89/	GP, ACT	NT, DDD, GAM, COV Rev. 4 (1999) available with slight changes below 20 MeV.
44	*29-Cu-66 5.1 m*	ADL-3	Jun94/	ACT	Only sets of cross-sections given.
45	*29-Cu-67 61.9 h*	ADL-3	Jun94/	ACT	Only sets of cross-sections given.
46	**30-Zn**	BROND-2	Dec89/Oct91	GP	NT, GAM
47	**30-Zn-64**	BROND-2	Dec89/Mar92	GP	NT, DD
48	*30-Zn-65 244.3 d*	ADL-3	Jan94/	ACT	Only sets of cross-sections given.
49	**30-Zn-66**	BROND-2	Dec89/	GP	NT, DD
50	**30-Zn-67**	BROND-2	Dec89/Feb92	GP	NT, DD
51	**30-Zn-68**	BROND-2	Dec89/Feb92	GP	NT, DD
52	**30-Zn-70**	BROND-2	Dec89/Feb92	GP	NT, DD
53	*30-Zn-71m 3.9 h*	ADL-3	Jan94/	ACT	Only sets of cross-sections given.
54	*30-Zn-72 46.5 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.

ELEMENTS IN INTERMEDIATE GROUP 1

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
1	**31-Ga**	FOND-2	Sep72/Dec90	GP	NT, GAM Modification of ENDL-72 evaluation.
2	**31-Ga-69**	JENDL-3.2	Mar94/	GP	NT
3	**31-Ga-71**	JENDL-3.2	Mar94/	GP	NT
4	**32-Ge**	JENDL-3.2	Mar94/	GP	NT
5	*32-Ge-68* *270.82 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
6	*32-Ge-69* *39.0 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
7	**32-Ge-70**	JENDL-3.2	Mar94/	FP	**NT**
8	*32-Ge-71* *11.43 d*	EAF-3	Jul92/Nov96	FP	MF=10 data added for MT=106.
9	**32-Ge-72**	JENDL-3.2	Mar94/	FP	NT
10	**32-Ge-73**	JENDL-3.2	Mar94/	FP	NT
11	**32-Ge-74**	JENDL-3.2	Mar94/	FP	NT
12	**32-Ge-76** **1.53×10^{21} y**	JENDL-3.2	Mar94/Jul99	FP	NT Threshold reaction cross-sections reviewed.
13	*33-As-71* *65.28 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
14	*33-As-72* *26.0 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
15	*33-As-73* *80.3 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
16	*33-As-74* *17.77 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
17	**33-As-75**	JENDL-3.2	Mar90/Feb94 /Jul99	GP	NT Threshold reaction cross-sections reviewed.
18	*33-As-76* *26.4 h*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
19	*33-As-77* *38.8 h*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
20	*34-Se-72* *8.5 d*	EAF-3	Jul92/Nov96	FP	MF=10 data added for MT=106.
21	*34-Se-73* *7.1 h*	EAF-3	Jul92/Nov96	FP	MF=10 data added for MT=104.
22	**34-Se-74**	JENDL-3.2	Mar90/Nov96	FP	NT MF=9 data added for MT=16.
23	*34-Se-75* *119.64d*	EAF-3	Jul92/Nov96	FP	Neutron data represented only by reaction cross-sections. MF=10 data added for MT=106.
24	**34-Se-76**	JENDL-3.2	Mar90/	FP	NT
25	**34-Se-77**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
26	**34-Se-78**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
27	*34-Se-79* *$6.\times10^{4}$ y*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
28	**34-Se-80**	JENDL-3.2	Mar93/Jul99	FP	NT Threshold reaction cross-sections reviewed.
29	**34-Se-82** **1.08×10^{20} y**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
30	*35-Br-77* *57.0 h*	ADL-3	Jan94/	FP	Only sets of cross-sections given.
31	**35-Br-79**	JENDL-3.2	Mar90/Mar93	FP	NT

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
32	**35-Br-81**	JENDL-3.2	Mar93/Jul99	FP	NT Threshold reaction cross-sections reviewed.
33	*35-Br-82* *35.34 h*	ADL-3	Jan94/Jul99	FP	Only sets of cross-sections given. Threshold reaction cross-sections reviewed.
34	**36-Kr-78**	JENDL-3.2	Mar90/	FP	NT
35	*36-Kr-79* *34.9 h*	ADL-3	Jan94/	FP	Only sets of cross-sections given.
36	**36-Kr-80**	JENDL-3.2	Mar90/Nov96	FP	NT Cross-sections modified. DD data added.
37	*36-Kr-81* *$2.3{\times}10^5$ y*	EAF-3	Jul92/Nov96	FP	Only sets of cross-sections given. DD data added.
38	**36-Kr-82**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
39	**36-Kr-83**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
40	**36-Kr-84**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
41	*36-Kr-85* *10.76 y*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
42	**36-Kr-86**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
43	*37-Rb-83* *86.2 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
44	*37-Rb-84* *32.8 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
45	**37-Rb-85**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
46	*37-Rb-86* *18.7 d*	JEF-2	Jul82/Jul99	FP	NT Modification of ENDF/B-5FP evaluation. Threshold reaction cross-sections reviewed.
47	**37-Rb-87**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
48	*38-Sr-82* *25.34 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
49	*38-Sr-83* *32.4 h*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
50	**38-Sr-84**	ENDF/B-5	Feb80/	FP	NT
51	*38-Sr-85* *64.9 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added. Threshold reaction cross-sections reviewed.
52	**38-Sr-86**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
53	**38-Sr-87**	JENDL-3.2	Mar90/	FP	NT
54	**38-Sr-88**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
55	*38-Sr-89* *50.5 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
56	38-Sr-90 *28.64 y*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
57	*39-Y-87* *80.3 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
58	*39-Y-88* *106.6 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
59	**39-Y-89**	JENDL-3.2	Nov93/Jul99	GP	NT, GAM Threshold reaction cross-sections reviewed.

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
60	39-Y-90 *64.1 h*	JEF-2	Jul82/Jul99	FP	NT Modification of ENDF/B-5FP evaluation. Threshold reaction cross-sections reviewed.
61	39-Y-91 *58.5 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
62	**40-Zr**	BROND-2	Dec88/Sep98	GP	NT, GAM Upper limit of unresolved resonance region changed for Zr-90 to 200 keV.
63	*40-Zr-88* *83.4 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
64	*40-Zr-89* *78.4 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
65	**40-Zr-90**	JENDL-3.2	Aug89/Jul99	FP	NT Threshold reaction cross-sections reviewed.
66	**40-Zr-91**	BROND-2	Dec88/Jul99	FP	NT Threshold reaction cross-sections reviewed.
67	**40-Zr-92**	JENDL-3.2	Aug89Jul99	FP	NT Threshold reaction cross-sections reviewed.
68	40-Zr-93 *1.5×10^{6} y*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
69	**40-Zr-94**	JENDL-3.2	Aug89/Jul99	FP	NT Threshold reaction cross-sections reviewed.
70	40-Zr-95 *64.0 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
71	**40-Zr-96** **3.9×10^{18} y**	JENDL-3.2	Aug89/Jul99	FP	NT Threshold reaction cross-sections reviewed.

ELEMENTS IN INTERMEDIATE GROUP 2

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
1	*41-Nb-91* *680 y*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
2	*41-Nb-91m* *60.9 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
3	*41-Nb-92* *3.6 $\times 10^7$ y*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
4	*41-Nb-92m* *10.15 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
5	**41-Nb-93**	BROND-2	Dec88/Feb93	GP	NT, GAM **Cross section for (n, 2n) reaction recommended as a standard.** Corresponds to the ENDF/B-6 data.
6	*41-Nb-93m* *16.13 y*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
7	*41-Nb-94* *2.$\times 10^4$ y*	JENDL-3.2	Mar90/Nov96	FP	NT Cross-sections modified. DD data added.
8	*41-Nb-95* *34.97 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
9	*41-Nb-95m* *86.6 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
10	*41-Nb-96* *23.4 h*	ADL-3	Jan94/	FP	Only sets of cross-sections given.
11	**42-Mo**	JENDL-3.2	Mar89/Feb94	GP	NT, GAM
12	**42-Mo-92**	JENDL-3.2	Aug89/Oct93	FP	NT
13	*42-Mo-93* *3.5$\times 10^3$ y*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
14	**42-Mo-94**	JENDL-3.2	Aug89/Oct93	FP	NT
15	**42-Mo-95**	JENDL-3.2	Aug89/Oct93/Jul99	FP	NT Threshold reaction cross-sections reviewed.
16	**42-Mo-96**	JENDL-3.2	Aug89/Oct93/Jul99	FP	NT Threshold reaction cross-sections reviewed.
17	**42-Mo-97**	JENDL-3.2	Aug89/Oct93/Jul99	FP	NT Threshold reaction cross-sections reviewed.
18	**42-Mo-98**	JENDL-3.2	Aug89/Oct93/Jul99	FP	NT Threshold reaction cross-sections reviewed.
19	*42-Mo-99* *66 h*	JENDL-3.2	Aug89/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
20	**42-Mo-100** **1.15$\times 10^{19}$ y**	JENDL-3.2	Aug89/Oct93/Jul99	FP	NT Threshold reaction cross-sections reviewed.
21	*43-Tc-95* *20 h*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
22	*43-Tc-95m* *60 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
23	*43-Tc-96* *4.3 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
24	*43-Tc-97* *4.$\times 10^6$ y*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
25	*43-Tc-97m* *92.2 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
26	*43-Tc-98* *4.2×10^6 y*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
27	*43-Tc-99* *2.1×10^5 y*	JENDL-3.2	Mar90/Nov93/Jul99	GP	NT Threshold reaction cross-sections reviewed.
28	**44-Ru-96**	JENDL-3.2	Mar90/	FP	NT
29	*44-Ru-97* *2.9 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
30	**44-Ru-98**	JENDL-3.2	Mar90/	FP	NT
31	**44-Ru-99**	JENDL-3.2	Mar90/Oct93	FP	NT
32	**44-Ru-100**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
33	**44-Ru-101**	JENDL-3.2	Mar90/Oct93/Jul99	FP	NT Threshold reaction cross-sections reviewed.
34	**44-Ru-102**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
35	*44-Ru-103* *39.35 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
36	**44-Ru-104**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
37	*44-Ru-105* *4.44 h*	EAF-3	Jul92/Nov96	FP	Neutron data represented only by reaction cross-sections. Cross-sections modified. DD data added.
38	*44-Ru-106* *373.6 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
39	*45-Rh-101* *3.3 y*	EAF-3	Jul92/Nov96	FP	Neutron data represented only by reaction cross-sections. Threshold reaction cross-sections reviewed.
40	*45-Rh-102m* *2.9 y*	EAF-3	Jul92/Nov96	FP	Neutron data represented only by reaction cross-sections. Threshold reaction cross-sections reviewed.
41	*45-Rh-102* *207. d*	EAF-3	Jul92/Nov96	FP	Neutron data represented only by reaction cross-sections. Threshold reaction cross-sections reviewed.
42	**45-Rh-103**	JENDL-3.2	Mar90/Feb94/Jul99	GP	NT Threshold reaction cross-sections reviewed.
43	*45-Rh-105* *35.4 h*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
44	**46-Pd-102**	JENDL-3.2	Mar90/	FP	NT ENDF/B-6 evaluation (1998) available.
45	*46-Pd-103* *16.96 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
46	**46-Pd-104**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation (1998) available.
47	**46-Pd-105**	JENDL-3.2	Mar90/Aug91/Jul99	FP	NT Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation (1998) available.
48	**46-Pd-106**	JENDL-3.2	Mar90/Jul99	FP	FP Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation (1998) available.
49	*46-Pd-107* *6.5×10^6 y*	JENDL-3.2	Mar90/Mar93/Jul99	FP	NT Threshold reaction cross-sections reviewed.
50	**46-Pd-108**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation (1998) available.

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
51	**46-Pd-110**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation (1998) available.
52	**47-Ag**	JENDL-3.2	Mar87/Feb94	GP	NT, GAM
53	*47-Ag-105g* *41.29 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
54	*47-Ag-106m* *8.3 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
55	**47-Ag-107**	JENDL-3.2	Mar87/Feb94	ACT	NT, GAM
56	*47-Ag-108m* *418 y*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
57	**47-Ag-109**	JENDL-3.2	Mar87/Feb94/J ul99	FP	NT, GAM Threshold reaction cross-sections reviewed.
58	*47-Ag-110m* *249.9 d*	JENDL-3.2	Mar90/	FP	NT
59	*47-Ag-111* *7.45 d*	JEF-2	Jul82/Jul99	FP	NT Modification of ENDF/B-5FP evaluation. Threshold reaction cross-sections reviewed.
60	**48-Cd**	JENDL-3.2	Mar89/Dec93	GP	NT, GAM
61	**48-Cd-106**	JENDL-3.2	Mar90/Nov96	FP	NT Cross-sections modified. DD data added.
62	**48-Cd-108**	JENDL-3.2	Mar90/	FP	NT Cross-sections modified. DD data added.
63	*48-Cd-109* *462 d*	EAF-3	Jul92//Nov96	FP	Cross-sections modified. DD data added.
64	**48-Cd-110**	JENDL-3.2	Mar90/Nov93/J ul99	FP	NT Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation, Rev. 3 (1993) available.
65	**48-Cd-111**	JENDL-3.2	Mar90/Nov93/J ul99	FP	NT Threshold reaction cross-sections reviewed.
66	**48-Cd-112**	JENDL-3.2	Mar90/Jul99	FP	FP Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation, Rev. 3 (1993) available.
67	**48-Cd-113** **9×10^{15} y**	JENDL-3.2	Mar90/Aug93/J ul99	FP	FP Threshold reaction cross-sections reviewed.
68	***48-Cd-113m*** ***14.6 y***	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
69	**48-Cd-114**	JENDL-3.2	Mar90/Nov96/J ul99	FP	FP DD data added. Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation, Rev. 3 (1993) available.
70	*48-Cd-115* *53.38 h*	EAF-3	Jul92/Nov96	FP	Only sets of cross-sections given.
71	*48-Cd-115m* *44.8 d*	JEF-2	Jul82/	FP	FP Modification of ENDF/B-5FP evaluation.
72	**48-Cd-116** **2.6×10^{19} y**	JENDL-3.2	Mar90/Nov96/J ul99	FP	NT DD data added. Threshold reaction cross-sections reviewed.

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
					ENDF/B-6 evaluation, Rev. 3 (1993) available.
73	**49-In-113**	JENDL-3.2	Mar90/	FP	NT
74	*49-In-114m 49.5 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
75	**49-In-115 4.4×10^{14} y**	JENDL-3.2	Mar90/Mar97/Jul99	FP	NT, DD, COV Threshold reaction cross-sections reviewed.
76	**50-Sn-112**	JENDL-3.2	Mar90/	FP	NT
77	*50-Sn-113 115.1 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
78	**50-Sn-114**	JENDL-3.2	Mar90/	FP	NT
79	**50-Sn-115**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
80	**50-Sn-116**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
81	**50-Sn-117**	JENDL-3.2	Mar90/Nov96/Jul99	FP	NT DD data added. Threshold reaction cross-sections reviewed.
82	*50-Sn-117m 13.6 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
83	**50-Sn-118**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
84	**50-Sn-119**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
85	*50-Sn-119m 293 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
86	**50-Sn-120**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
87	*50-Sn-121 27.0 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
88	*50-Sn-121m 50 y*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
89	**50-Sn-122**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
90	*50-Sn-123 129.3 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
91	**50-Sn-124**	JENDL-3.2	Mar90/Mar93/Jul99	FP	NT Threshold reaction cross-sections reviewed.
92	*50-Sn-125 9.64 d*	JEF-2	Jul82/Jul99	FP	NT Modification of ENDF/B-5FP evaluation. Threshold reaction cross-sections reviewed.
93	*50-Sn-126 10^{5} y*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.

ELEMENTS IN INTERMEDIATE GROUP 3

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
1	**51-Sb**	JENDL-3.2	Mar89/Feb94	GP	NT
2	*51-Sb-119* *38.5 h*	EAF-3	JuL92/Nov96	FP	Cross-sections modified. DD data added.
3	*51-Sb-120m* *5.76 d*	EAF-3	JuL92/Nov96	FP	Cross-sections modified. DD data added.
4	**51-Sb-121**	JENDL-3.2	Mar89/Feb94/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
5	*51-Sb-122* *2.70 d*	EAF-3	Jul92/Nov96	FP	Threshold reaction cross-sections reviewed.
6	**51-Sb-123**	JENDL-3.2	Aug89/Feb94/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
7	*51-Sb-124* *60.3 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
8	*51-Sb-125* *2.77 y*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
9	*51-Sb-126* *12.4 d*	JEF-2	Jul82/Jul99	FP	NT Modification of ENDF/B-5FP evaluation. Threshold reaction cross-sections reviewed.
10	*51-Sb-127* *3.85 d*	EAF-3	Jul92/Jul99	FP	NT, DD Threshold reaction cross-sections reviewed.
11	*52-Te-118* *6.0 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
12	*52-Te-119m* *4.7 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
13	**52-Te-120**	JENDL-3.2	Mar90/	FP	NT
14	*52-Te-121* *16.8 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
15	*52-Te-121m* *154 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
16	**52-Te-122**	JENDL-3.2	Mar90/Nov93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
17	**52-Te-123** **1.24×10^{13} y**	JENDL-3.2	Mar90/Sep93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
18	*52-Te-123m* *119.7 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
19	**52-Te-124**	JENDL-3.2	Mar90/Nov93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
20	**52-Te-125**	JENDL-3.2	Mar90/Nov93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
21	*52-Te-125m* *57.4 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
22	**52-Te-126**	JENDL-3.2	Mar90/Nov93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
23	*52-Te-127m* *109 d*	JENDL-3.2	Mar90/	FP	NT
24	**52-Te-128** **7.2×10^{24} y**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
25	*52-Te-129m* *33.6 d*	JENDL-3.2	Mar90/	FP	NT
26	**52-Te-130** **2.7×10^{21} y**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
27	*52-Te-131m* *30 h*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
28	*52-Te-132* *76.3 h*	JEF-2	Jul82/Jul99	FP	NT Modification of ENDF/B-5FP evaluation. Threshold reaction cross-sections reviewed.
29	*53-I-124* *4.15 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
30	*53-I-125* *59.41 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
31	*53-I-126* *13.11 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
32	**53-I-127**	JENDL-3.2	Mar90/Apr93/ Jul99	GP	NT Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation (1993) available.
33	*53-I-128* *25 m*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
34	*53-I-129* *$1.57{\times}10^7$ y*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
35	*53-I-130* *12.36 h*	JEF-2	Jul82/Jul99	FP	NT Modification of ENDF/B-5FP evaluation. Threshold reaction cross-sections reviewed.
36	*53-I-131* *8.02 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
37	*53-I-135* *6.61 h*	ADL-3	Jan94/	FP	Only sets of cross-sections given.
38	**54-Xe-124**	JENDL-3.2	Mar90/	FP	NT
39	**54-Xe-126**	JENDL-3.2	Mar90/	FP	NT
40	54-Xe-127 *36.4 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
41	**54-Xe-128**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
42	**54-Xe-129**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
43	54-Xe-129m *8.89 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
44	**54-Xe-130**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
45	**54-Xe-131**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
46	54-Xe-131m *11.9 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
47	**54-Xe-132**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
48	*54-Xe-133* *5.25 d*	ENDF/B-6	Apr74/Jul99	FP	NT Threshold reaction cross-sections reviewed.
49	*54-Xe-133m* *2.19 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
50	**54-Xe-134**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
51	*54-Xe-135* *9.10 h*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
52	**54-Xe-136**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
53	*55-Cs-131* *9.69 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
54	*55-Cs-132* *6.47 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
55	**55-Cs-133**	JENDL-3.2	Mar90/Jul99	GP	NT Threshold reaction cross-sections reviewed.
56	*55-Cs-134* *2.06 y*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
57	*55-Cs-135* *2×10^6 y*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
58	*55-Cs-136* *13.16 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
59	*55-Cs-137* *30.17 y*	JENDL-3.2	Mar90/Mar93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
60	**56-Ba-130**	JENDL-3.2	Mar90/	FP	NT
61	*56-Ba-131* *11.5 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
62	**56-Ba-132**	JENDL-3.2	Mar90/	FP	NT
63	*56-Ba-133* *10.5 y*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
64	*56-Ba-133m* *38.9 h*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
65	**56-Ba-134**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
66	**56-Ba-135**	JENDL-3.2	Mar90/Nov93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
67	*56-Ba-135m* *28.7 h*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
68	**56-Ba-136**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
69	**56-Ba-137**	JENDL-3.2	Mar90/Oct93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
70	**56-Ba-138**	JENDL-3.2	Mar90/Oct93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
71	*56-Ba-139* *83.06 m*	EAF-3	Jul92/Nov96/ Jul99	FP	Cross-sections modified. DD data added.
72	*56-Ba-140* *12.75 d*	JEF-2	Jul82/Jul99	FP	NT Modification of ENEA/CEA evaluation. Threshold reaction cross-sections reviewed.
73	*57-La-137* *$6.\times10^4$ y*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
74	**57-La-138** **1.05×10^{11} y**	JENDL-3.2	Mar90/Nov93	GP	NT
75	**57-La-139**	JENDL-3.2	Mar90/Nov93/ Jul99	GP	NT Threshold reaction cross-sections reviewed.
76	*57-La-140* *40.272 h*	EAF-3	Jul92/Nov96/ Jul99	FP	DD data added. Threshold reaction cross-sections reviewed.
77	**58-Ce-136**	EAF-3	Jul92/	FP	Only sets of cross-sections given.
78	*58-Ce-137m* *34.4 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
79	**58-Ce-138**	EAF-3	Jul92/	FP	Only sets of cross-sections given.
80	*58-Ce-139* *137.6 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
81	**58-Ce-140**	JENDL-3.2	Mar90/Nov93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
82	*58-Ce-141* *32.50 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
83	**58-Ce-142**	JENDL-3.2	Mar90/Sep93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
84	*58-Ce-143* *33.0 h*	EAF-3	Jul92/Nov96/ Jul99	FP	DD data added. Threshold reaction cross-sections reviewed.
85	*58-Ce-144* *284.8 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
86	**59-Pr-141**	JENDL-3.2	Mar90/Sep93/ Jul99	GP	NT Threshold reaction cross-sections reviewed.
87	*59-Pr-142* *19.13 h*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
88	*59-Pr-143* *13.75 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
89	*60-Nd-140* *3.37 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
90	*60-Nd-141* *2.5 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
91	**60-Nd-142**	JENDL-3.2	Mar90/Nov93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
92	**60-Nd-143**	JENDL-3.2	Mar90/Nov93/ Jul99	FP	NT Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation, 1993 Revision, available for thermal and resonance energy region.
93	**60-Nd-144** **2.29×10^{15} y**	JENDL-3.2	Mar90/Oct93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
94	**60-Nd-145**	JENDL-3.2	Mar90/Oct93/ Jul99	FP	NT Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation, 1993 Revision, available for thermal and resonance energy region.
95	**60-Nd-146**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
96	*60-Nd-147* *10.98 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation, 1993 Revision, available for thermal and resonance energy region.
97	**60-Nd-148**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
98	**60-Nd-150** **1.4×10^{19} y**	JENDL-3.2	Mar90/Oct93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.

ELEMENTS IN INTERMEDIATE GROUP 4

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
1	*61-Pm-143* *265 d*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
2	*61-Pm-144* *1.0 y*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
3	*61-Pm-145* *17.7 y*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
4	*61-Pm-146* *5.53 y*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
5	*61-Pm-147* *2.62 y*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
6	*61-Pm-148* *5.37 d*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
7	*61-Pm-148m* *41.3 d*	JENDL-3.2	Mar90/Dec97	FP	NT
8	*61-Pm-149* *53.1 h*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
9	*61-Pm-150* *2.7 h*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
10	*61-Pm-151* *28.4 h*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
11	**62-Sm-144**	ENDF/B-6	Feb80/Jan92	FP	NT
12	*62-Sm-145* *340 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
13	*62-Sm-146* *1.03×10^{8} y*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
14	**62-Sm-147** **1.06×10^{11} y**	FOND-2.2	Mar90/Aug93/ Jul99	FP	NT Modification of JENDL-3.2 evaluation. Threshold reaction cross-sections reviewed.
15	**62-Sm-148** **7×10^{15} y**	FOND-2.2	Sep87/Apr97/ Jul99	FP	NT Modification of JENDL-3.2 evaluation. Threshold reaction cross-sections reviewed.
16	**62-Sm-149**	FOND-2.2	Mar90/Jul99/ Aug99	FP	NT Modification of JENDL-3.2 evaluation. Resonance parameters taken from BROND-2 for unresolved resonance region. Threshold reaction cross-sections reviewed.
17	**62-Sm-150**	JENDL-3.2	Mar90/Jun94/ Jul99	FP	NT Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation, 1993 Revision, available for thermal and resonance energy region.
18	*62-Sm-151* *93 y*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
19	**62-Sm-152**	JENDL-3.2	Mar90/Jun94/ Jul99	FP	NT Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation, 1993 Revision, available for thermal and resonance energy region.
20	*62-Sm-153* *47.26 h*	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
21	**62-Sm-154**	JENDL-3.2	Mar90/Jun94/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
22	**63-Eu**	JENDL-3.2	Mar89/Nov93	GP	NT, GAM
23	*63-Eu-145 5.93 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
24	*63-Eu-146 4.51 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
25	*63-Eu-147 24.6 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
26	*63-Eu-148 55.6 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
27	*63-Eu-149 93.1 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
28	*63-Eu-150 36.9 y*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
29	**63-Eu-151**	JENDL-3.2	Mar89/	FP	NT
30	*63-Eu-152 13.33 y*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
31	**63-Eu-153**	JENDL-3.2	Mar89/Jan94/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
32	*63-Eu-154 8.8 y*	JENDL-3.2	Dec90/Nov93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
33	*63-Eu-155 4.761 y*	JENDL-3.2	Mar90/Nov93/ Jul99	FP	NT Threshold reaction cross-sections reviewed.
34	*63-Eu-156 15.2 d*	JEF-2	Jul82/Apr89/ Jul99	FP	NT Modification of ENDF/B-5FP evaluation. Threshold reaction cross-sections reviewed.
35	*63-Eu-157 15.18 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
36	*64-Gd-148 74.6 y*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
37	*64-Gd-149 9.28 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
38	*64-Gd-150* 1.8×10^{6} *y*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
39	*64-Gd-151 120 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
40	**64-Gd-152** $\mathbf{1.1\times10^{14}}$ **y**	JENDL-3.2	Mar90/	FP	NT ENDF/B-6 evaluation (1997) available.
41	*64-Gd-153 239.47 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
42	**64-Gd-154**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed. ENDF/B-6 evaluation (1997) available.
43	**64-Gd-155**	ENDF/B-6	Jun77/Jul99	FP	NT Threshold reaction cross-sections reviewed.
44	**64-Gd-156**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
45	**64-Gd-157**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
46	**64-Gd-158**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.
47	**64-Gd-160**	JENDL-3.2	Mar90/Jul99	FP	NT Threshold reaction cross-sections reviewed.

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
48	*65-Tb-157* *99 y*	AF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
49	*65-Tb-158* *180 y*	EAF-3	Jul92/Nov96	FP	Cross-sections modified. DD data added.
50	**65-Tb-159**	JENDL-3.2	Mar90/Jul99	GP	NT Threshold reaction cross-sections reviewed.
51	*65-Tb-160* *73.3 d*	JEF-2	Jun82/Jul99	FP	NT Modification of ENDF/B-5FP evaluation. Threshold reaction cross-sections reviewed.
52	*65-Tb-161* *6.90 d*	EAF-3	Jul92/Jul99	FP	Only sets of cross-sections given. Threshold reaction cross-sections reviewed.
53	**66-Dy**	FOND-2	Nov89/	GP	Only set of resonance parameters given.
54	*66-Dy-154* *3.0×10^{6} y*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
55	*66-Dy-155* *10.0 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
56	**66-Dy-156**	EAF-3	Jul92/	FP	Only sets of cross-sections given.
57	**66-Dy-158**	EAF-3	Jul92/	FP	Only sets of cross-sections given.
58	*66-Dy-159* *144.4 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
59	**66-Dy-160**	JEF-2	Jun82/Jul99	FP	NT Modification of ENDF/B-5FP evaluation. Threshold reaction cross-sections reviewed.
60	**66-Dy-161**	JEF-2	Jun82/Jul99	FP	NT Modification of ENDF/B-5FP evaluation. Threshold reaction cross-sections reviewed.
61	**66-Dy-162**	JEF-2	Jun82/Jul99	FP	NT Modification of ENDF/B-5FP evaluation. Threshold reaction cross-sections reviewed.
62	**66-Dy-163**	JEF-2	Jun82/Jul99	FP	NT Modification of ENDF/B-5FP evaluation. Threshold reaction cross-sections reviewed.
63	**66-Dy-164**	JEF-2	Jun82/Jul99	FP	NT Modification of ENDF/B-5FP evaluation. Threshold reaction cross-sections reviewed.
64	*66-Dy-165* *2.35 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
65	*66-Dy-166* *81.5 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
66	*67-Ho-163* *4750 y*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
67	**67-Ho-165**	ENDF/B-5	Apr74/Jul80	GP	NT ENDF/B-6 evaluation (1998) available.
68	*67-Ho-166* *26.8 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
69	*67-Ho-166m* *1 200 y*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
70	**68-Er**	FOND-2	Dec76/Nov89	GP	Only set of resonance parameters for stable isotopes given.
71	**68-Er-162**	BROND-2	Dec76/Nov89	FP	NT, DD
72	**68-Er-164**	BROND-2	Dec76/Nov89	FP	NT, DD

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
73	*68-Er-165 10.3 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
74	**68-Er-166**	BROND-2	Dec76/Nov89	FP	NT, DD
75	**68-Er-167**	BROND-2	Dec76/Nov89	FP	NT, DD
76	**68-Er-168**	BROND-2	Dec76/Nov89	FP	NT, DD
77	*68-Er-169 9.40 d*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
78	**68-Er-170**	BROND-2	Dec76/Nov89	FP	NT, DD
79	*68-Er-171 7.52 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
80	*68-Er-172 49 h*	EAF-3	Jul92/	FP	Only sets of cross-sections given.
81	*69-Tm-167 9.25 d*	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.
82	*69-Tm-168 93.1 d*	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.
83	**69-Tm-169**	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.
84	*69-Tm-170 128.6 d*	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.
85	*69-Tm-171 1.92 y*	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.
86	*69-Tm-172 63.6 h*	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.
87	**70-Yb-168**	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.
88	*70-Yb-169 32.0 d*	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.
89	**70-Yb-170**	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.
90	**70-Yb-171**	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.
91	**70-Yb-172**	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.
92	**70-Yb-173**	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.
93	**70-Yb-174**	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.
94	*70-Yb-175 4.2 d*	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.
95	**70-Yb-176**	EAF-3	Jul92/	FP	Neutron data represented only by reaction cross-sections.

ELEMENTS IN INTERMEDIATE GROUP 5

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
1	*71-Lu-173 1.37 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
2	*71-Lu-174 3.31 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
3	*71-Lu-174m 142 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
4	**71-Lu-175**	ENDF/B-5	Jul67/	GP	NT
5	**71-Lu-176 3.8×10^{10} y**	ENDF/B-5	Jul67/	GP	NT
6	*71-Lu-177 6.71 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
7	*71-Lu-177m 160.1 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
8	**72-Hf**	ENDL-82	Oct82/Dec92	GP	NT, GP
9	*72-Hf-172 1.87 y*	ADL-3	Jan94/	ACT	Only sets of cross-sections given.
10	*72-Hf-173 23.6 h*	ADL-3	Jan94/	ACT	Only sets of cross-sections given.
11	**72-Hf-174 2.0×10^{15} y**	JENDL-3	Dec82/Dec83	GP	NT ENDF/B-6 evaluation (1993) available.
12	*72-Hf-175 70.0 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
13	**72-Hf-176**	JENDL-3	Dec82/Dec83	GP	NT ENDF/B-6 evaluation (1993) available.
14	**72-Hf-177**	JENDL-3	Dec82/Dec83	GP	NT ENDF/B-6 evaluation (1993) available.
15	**72-Hf-178**	JENDL-3	Dec82/Jan84	GP	NT ENDF/B-6 evaluation (1993) available.
16	*72-Hf-178n 31 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
17	**72-Hf-179**	JENDL-3	Dec82/Dec83	GP	NT ENDF/B-6 evaluation (1993) available.
18	*72-Hf-179n 25 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
19	**72-Hf-180**	JENDL-3	Dec82/Dec83	GP	NT ENDF/B-6 evaluation (1993) available.
20	*72-Hf-181 42.39 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
21	*72-Hf-182 9×10^{6} y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
22	*73-Ta-179 665 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
23	**73-Ta-180 8.15 h**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
24	**73-Ta-180m $>10^{15}$ y**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
25	**73-Ta-181**	ENDL-72	Jan72/Dec92	GP	NT, GAM
26	*73-Ta-182 114.43 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
27	*73-Ta-183 5.0 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
28	*74-W-178 22 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
29	**74-W-180**	BROND-2	May83/May90	ACT	Only set of resonance parameters given.
30	*74-W-181* *121.2 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
31	**74-W-182**	BROND-2	May83/May90	ACT	NT, GAM
32	**74-W-183**	BROND-2	May83/May90	ACT	NT, GAM
33	**74-W-184**	BROND-2	May83/May90	ACT	NT, GAM
34	*74-W-185* *75.1 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
35	**74-W-186**	BROND-2	May83/May90	ACT	NT, GAM
36	*74-W-187* *23.72 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
37	*74-W-188* *69 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
38	**75-Re**	FOND-2	Jan88/Sep91	GP	NT
39	*75-Re-183* *71 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
40	*75-Re-184* *38.0 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
41	*75-Re-184m* *169 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
42	**75-Re-185**	FOND-2	Jan68/Dec92	ACT	NT Modification of ENDF/B-5 evaluation.
43	*75-Re-186* *89.25 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
44	*75-Re-186m* *2×10^5 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
45	**75-Re-187** **5×10^{10} y**	FOND-2	Jan68/Dec92	ACT	NT Modification of ENDF/B-5 evaluation.
46	*75-Re-188* *16.96 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
47	*75-Re-189* *24.3 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
48	**76-Os**	BROND-2	Jan90/	GP	NT, GP
47	**76-Os-184**	FOND-2	Jul92/	ACT	NT, DD
50	*76-Os-185* *94 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
51	**76-Os-186** **2×10^{15} y**	FOND-2	Jul92/	ACT	NT, DD
52	**76-Os-187**	FOND-2	Jul92/	ACT	NT, DD
53	**76-Os-188**	FOND-2	Jul92/	ACT	NT, DD
54	**76-Os-189**	FOND-2	Jul92/	ACT	NT, DD
55	**76-Os-190**	FOND-2	Jul92/	ACT	NT, DD
56	*76-Os-191* *15.4 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
57	**76-Os-192**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
58	*76-Os-193* *30.11 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
59	*76-Os-194* *6.0 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
60	**77-Ir**	BROND-2	Jan90/	GP	NT, GAM
61	*77-Ir-189* *13.3 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
62	*77-Ir-190* *11.8 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
63	**77-Ir-191**	FOND-2	Jan90/Nov93	ACT	NT, DD
64	*77-Ir-192* *73.83 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
65	*77-Ir-192n* *241 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
66	**77-Ir-193**	FOND-2	Jan90/Jan93	ACT	NT, DD
67	*77-Ir-193m* *10.53 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
68	*77-Ir-194m* *171 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
69	**78-Pt**	ENDL-78	Oct82/Apr91	GP	NT, GP
70	**78-Pt-190** **6.5×10^{11} y**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
71	*78-Pt-191* *2.8 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
72	**78-Pt-192**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
73	*78-Pt-193* *50 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
74	*78-Pt-193m* *4.33 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
75	**78-Pt-194**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
76	**78-Pt-195**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
77	*78-Pt-195m* *4.02 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
78	**78-Pt-196**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
79	*78-Pt-197* *18.3 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
80	**78-Pt-198**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
81	*79-Au-195* *196.1 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
82	*79-Au-196* *6.2 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
83	**79-Au-197**	ENDF/B-6 Rev. 1	Jan84/Jul91	GP ACT	NT, DD, GAM, COV **Cross section for the (n,γ) reaction in the 0.2-2.5 MeV region recommended as a standard.**
84	*79-Au-198* *2.6943 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
85	*79-Au-198m* *2.30 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
86	*79-Au-199* *3.139 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
87	*80-Hg-194* *520 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
88	*80-Hg-195* *9.5 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
89	*80-Hg-195m* *40 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
90	**80-Hg-196**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
91	*80-Hg-197* *64.1 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
92	*80-Hg-197m* *23.8 h*	ADL-3	Jan94/	ACT	Only sets of cross-sections given.

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
93	**80-Hg-198**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
94	**80-Hg-199**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
95	**80-Hg-200**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
96	**80-Hg-201**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
97	**80-Hg-202**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
98	*80-Hg-203* *46.59 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
99	**80-Hg-204**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.

ELEMENTS IN INTERMEDIATE GROUP 6

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
1	*81-Tl-201* *73.1 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
2	*81-Tl-202* *12.23 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
3	**81-Tl-203**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
4	*81-Tl-204* *3.78 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
5	**81-Tl-205**	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
6	**82-Pb**	JENDL-3.0	Jul87/	GP	NT, GAM
7	*82-Pb-202* *5.25×10^4 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
8	*82-Pb-203* *51.9 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
9	**82-Pb-204**	JENDL-3.0	Jul87/	ACT	NT, GAM
10	*82-Pb-205* *1.5×10^7 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
11	**82-Pb-206**	JENDL-3.0	Jul87/	ACT	NT, GAM
12	**82-Pb-207**	JENDL-3.0	Jul87/	ACT	NT, GAM
13	**82-Pb-208**	JENDL-3.0	Jul87/	ACT	NT, GAM
14	*82-Pb-209* *3.253 h*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
15	*82-Pb-210* *22.3 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given. Leads to the formation of Bi-210.
16	*83-Bi-207* *31.55 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
17	*83-Bi-208* *3.68×10^5 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
18	**83-Bi-209**	BROND-2	Nov90/	GP	NT, GAM
19	*83-Bi-210* *5.013 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
20	*83-Bi-210m* *3.0×10^6 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
21	*84-Po-208* *2.898 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
22	*84-Po-209* *102 y*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
23	*84-Po-210* *138.38 d*	EAF-3	Jul92/	ACT	Only sets of cross-sections given.
24	*88-Ra-223* *11.43 d*	JENDL-3.2	Aug88/	ACT	NT
25	*88-Ra-224* *3.66 d*	JENDL-3.2	Aug88/	ACT	NT
26	*88-Ra-225* *14.8 d*	JENDL-3.2	Aug88/	ACT	NT
27	*88-Ra-226* *1600 y*	JENDL-3.2	Aug88/Nov93	ACT	NT

ACTINIDES

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
1	*89-Ac-225* *10.0 d*	JENDL-3.2	Aug88/	MA	NT
2	*89-Ac-226* *29 h*	JENDL-3.2	Aug88/	MA	NT
3	*89-Ac-227* *21.773 y*	JENDL-3.2	Aug88/	MA	NT
4	*90-Th-227* *18.72 d*	JENDL-3.2	Aug88/Jun94	MA	NT
5	*90-Th-228* *1.913 y*	JENDL-3.2	Jun87/Jun94	MA	NT
6	*90-Th-229* *7 880 y*	JENDL-3.2	Aug88/Jun94	MA	NT
7	*90-Th-230* 7.54×10^{4} *y*	JENDL-3.2	Jul87/Jun94	MA	NT
8	**90-Th-232** 1.405×10^{10} *y*	ENDF/B-6	Dec77/	GP	NT, GAM, COV
9	*90-Th-233* *22.3 m*	JENDL-3.2	Jul87/Jun94	MA	NT
10	*90-Th-234* *24.10 d*	JENDL-3.2	Jul87/Jun94	MA	NT
11	*91-Pa-231* 3.276×10^{4} *y*	FOND-2.2	Mar87/Aug96	GP	NT Modification of capture cross-section for JENDL-3.2 evaluation.
12	*91-Pa-232* *1.31 d*	JENDL-3.2	Aug88/Jun94	MA	NT
13	*91-Pa-233* *27.0 d*	FOND-2	May78/	GP	NT, Modification of ENDF/B-5 evaluation.
14	*92-U-232* *68.9 y*	JENDL-3.2	Mar87/Mar94	GP	NT
15	*92-U-233* 1.592×10^{5} *y*	BROND-2	Mar90/Jun90	GP	NT
16	**92-U-234** 2.455×10^{5} *y*	FOND-2	Jul78/	GP	NT, DD Modification of ENDF/B-5 evaluation.
17	**92-U-235** 7.038×10^{8} *y*	ENDF/B-6 Rev. 2	Nov89/Feb93	GP	NT, GAM, COV **Cross-section for the (n, fis) reaction in the 100 keV-20 MeV energy region recommended as a standard.** ENDF/B-6 (Rev. 5 1998) reviews the parameters in the resonance energy region.
18	*92-U-236* 2.342×10^{7} *y*	ENDF/B-6	Oct89/	GP	NT, DD
19	*92-U-237* *6.72 d*	JENDL-3.2	Mar93/	MA	NT
20	**92-U-238** 4.468×10^{9} *y*	BROND-2	Jan80/Feb93	GP	NT, DD **Cross-section for the (n, fis) reaction in the energy region up to 20 MeV recommended as a standard.**
21	*93-Np-236* 1.54×10^{5} *y*	JENDL-3.2	Mar93/	MA	NT
22	*93-Np-237* 2.144×10^{6} *y*	JENDL-3.2	Nov87/Apr00	GP	NT, DD Modification of the data for the (n,3n) reaction. New BROND-3 (1996) evaluation available.

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
23	*93-Np-238* *2.117 d*	JENDL-3.2	Mar93/	MA	NT
24	*93-Np-239* *2.355 d*	ENDF/B-6	Dec88/	GP	NT
25	*94-Pu-236* *2.858 y*	ENDF/B-6	Apr78/	MA	NT, DD
26	*94-Pu-237* *45.2 d*	ENDF/B-6	Apr78/	MA	NT, DD ENDF/B-6 (Revision 1) adopts JENDL-3.2 evaluation modified in 1995.
27	*94-Pu-238* *87.74 y*	BROND-2	Feb87/Feb93	MA	NT
28	*94-Pu-239* 2.411×10^4 *y*	FOND-2	Mar87/Nov98	GP	NT, GAM Modification of JENDL-3 evaluation: data from the LIPAR-5 library adopted in the energy region below 200 eV.
29	*94-Pu-240* *6 563 y*	FOND-2	Dec80/Jan93	GP	NT, GAM Modification of BROND-2 evaluation.
30	*94-Pu-241* *14.35 y*	FOND-2	Feb79/Jan93	GP	NT, GAM Modification of BROND-2 evaluation.
31	*94-Pu-242* 3.750×10^5 *y*	BROND-2	Dec80/Apr91	GP	NT, GAM
32	*94-Pu-243* *4.956 h*	ENDF/B-6	Jul76/	MA	NT, DD, GAM
33	*94-Pu-244* 8.00×10^7 *y*	ENDF/B-6	Apr78/	MA	NT, DD
34	*95-Am-241* *432.2 y*	BROND-3	Feb97/Apr00	GP	NT, DD, GAM, COV Correction made to isomeric ratio for capture cross-section.
35	*95-Am-242* *16 h*	JENDL-3.2	Mar80/	GP	NT
36	*95-Am-242m* *141 y*	BROND-2	Oct90/Dec98	GP	NT Modification of BROND-2 evaluation.
37	*95-Am-243* *7 370 y*	JENDL-3.2	Mar88/	GP	NT BROND-3 evaluation (1997) and ENDF/B-6 evaluation (1998) available.
38	*95-Am-244* *10.1 h*	JENDL-3.2	Mar88/	MA	NT
39	*95-Am-244m* *26 m*	JENDL-3.2	Mar88/	MA	NT
40	*96-Cm-241* *32.8 d*	JENDL-3.2	Mar89/	MA	NT
41	*96-Cm-242* *162.94 d*	JENDL-3.2	Mar89/	MA	NT BROND-3 (1997) evaluation available.
42	*96-Cm-243* *29.1 y*	JENDL-3.2	Mar89/	MA	NT BROND-3 (1997) evaluation available.
43	*96-Cm-244* *18.10 y*	JENDL-3.2	Mar89/	MA	NT BROND-3 (1997) evaluation available.
44	*96-Cm-245* *8 500 y*	JENDL-3.2	Mar89/Sep92	MA	NT
45	*96-Cm-246* *4 730 y*	JENDL-3.2	Mar87/	MA	NT
46	*96-Cm-247* 1.56×10^7 *y*	JENDL-3.2	Mar89/	MA	NT
47	*96-Cm-248* 3.40×10^5 *y*	JENDL-3.2	Mar84/	MA	NT

No.	NUCLIDE	SOURCE	EVAL/REV	STATUS	Composition of data and brief comments
48	*96-Cm-249* *64.15 m*	JENDL-3.2	Mar84/Sep92	MA	NT
49	*96-Cm-250* *~9700 y*	JENDL-3.2	Aug87/Jul94	MA	NT
50	*97-Bk-249* *320 d*	ENDF/B-6	Jun86/	MA	NT
51	*97-Bk-250* *3.217 h*	JENDL-3.2	Mar87/	MA	NT
52	*98-Cf-249* *350.6 y*	ENDF/B-6	Apr89/	MA	NT
53	*98-Cf-250* *13.08 y*	ENDF/B-6	Jul76/	MA	NT, DD, GAM
54	*98-Cf-251* *898 y*	ENDF/B-6	Jul76/	MA	NT, DD, GAM
55	*98-Cf-252* *2.645 y*	ENDF/B-6	Jul76/Aug91	MA	NT, DD, GAM
56	*98-Cf-253* *17.81 d*	ENDF/B-6	Dec75	MA	NT
57	*98-Cf-254* *60.5 d*	JENDL-3.2	Aug87/Jun94	MA	NT
58	*99-Es-253* *20.47 d*	ENDF/B-6	Jul76/	MA	NT
59	*99-Es-254* *275.7 d*	JENDL-3.2	Aug87/Jun94	MA	NT
60	*99-Es-255* *39.8 d*	JENDL-3.2	Aug87/Jun94	MA	NT
61	*100-Fm-255* *20.1 h*	JENDL-3.2	Aug87/Jun94	MA	NT

Availability of data

The data are available from the Sectoral Fund for Algorithms and Programs at the following address:

Institute for Physics and Power Engineering, Sectoral Fund for Algorithms and Programs, Bondarenko Square 1, Kaluga Region, 249033, Russia
(Web: http//ultra.ippe.obninsk.ru:8097;
E-mail: ofap@ippe.rssi.ru, with copy of the request to abbn@ippe.rssi.ru)

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LIBRARIES OF DECAY DATA AND FISSION PRODUCT YIELDS IN THE ABBN-93 CONSTANT SET

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LIBRARIES OF DECAY DATA AND FISSION PRODUCT YIELDS IN THE ABBN-93 CONSTANT SET. This paper describes three new libraries in the ABBN constant set which are essential for calculating radioactivity: basic decay data, radioactive decay photon spectra and fission product yields.

1. Basic decay data

Decay data for radionuclides are needed in the ABBN-93 constant set primarily so that it can be used to calculate the nuclide composition and radiative characteristics of irradiated fuel and other reactor materials. Decay data are also needed to calculate delayed energy release components and photon formation in neutron reactions.

At this point in time there are several quite extensive compilations of decay data which are fairly comprehensively covered in the ENDF/B-6, JENDL-3.2 and JEF-2.2 evaluated nuclear data libraries. We should also mention the decay data libraries included in programs for calculating the nuclide composition of irradiated fuel and reactor materials - such as ORIGEN, Ref. [1], and FISPACT, Ref. [2]. One of the most extensive sources of information on decay data is the ENSDF (Evaluated Nuclear Structure Data File) international library, which contains both recommended and initial experimental data. There are several handbooks which are important sources of evaluated decay data: in particular, Ref. [3], which gives the spectra of photons irradiated by radionuclides taking into account internal conversion and other processes connected with the electron orbit of the atomic nucleus; and Ref. [4], which contains the branching ratios that lead, in particular, to the emission of delayed neutrons.

It is difficult to choose which data to use in practical calculations owing to the variety of information sources, their incompleteness and their evaluation inconsistency. Thus, choosing a standard set of decay data was in fact of the utmost importance in devising a set of constants laying claim to universality and standardization. It should be pointed out that the decay data compiled for ABBN-93 were not evaluated independently on the basis of initial experimental information. The main focus was on making the information provided as complete as possible, internally consistent and user-friendly in performing calculations in conjunction with ABBN-93 neutron and photon data.

The formats for the basic decay data are described in detail in Ref. [5]. The tables contain data for all the isotopes of a specific element, the symbol for which is indicated in the first line (see example below). At least one line is given to each isotope; a whole number identifier is given in the first column.

The first real parameter in the header line is the atomic number Z.

1st column -	four-digit isotope identifier: the three leftmost digits are the mass number; the rightmost digit is the isomer and stability identifier.
2nd column -	a letter indicating the nature of the content of the data given: a hyphen signifies that the isotope is radioactive and its values are given in the subsequent columns as listed below:
"A" signifies,	that the isotope is stable and, instead of the isomeric transition (IT) probability, the isotopic abundance in its natural mixture is given;
"B" signifies,	that as a result of at least one of the possible modes of decay, the product-nucleus may be formed not only in the ground state but also in a metastable state (in that case the data for this isotope are entered on two lines: whereas the decay probability is given, as for all radioactive isotopes, in the first line, the probability of a metastable state forming for the corresponding decay path is given in the second line);
"C" signifies,	that two isomeric states may form during decay, and three lines of data are given for the isotope under consideration; the probability of a first metastable state forming is given in the second line, and the probability of a second metastable state forming is given in the third line.
3rd column -	the unit of time for the half-life: s = seconds; m = minutes; d = days; y = years.
4th column -	the half-life in the unit indicated.
5th column -	"BETA+" is the (β^{+}+ε) decay probability.
6th column -	"BETA-" is the β^{-} decay probability.
7th column -	"IT" is the isomeric transition probability.
8th column -	"ALFA" is the alpha decay probability.
9th column -	"SF" is the spontaneous fission probability.
10th column -	"E-GAM " is the energy removed by photons.
11th column -	"E-BET " is the energy removed by electrons.
12th column -	"E-ALF" is the energy removed by alpha-particles and other heavy charged decay products (for example, fragments during spontaneous fission, protons emitted in several cases following beta-decay, etc.).
13th column -	indicates that there is a 15-group representation of the photon spectrum in the ABBN library of photon spectra for radioactive nuclei.

The MF=90 with MT=0 tables contain data on the energies released directly during decay. In addition, MF=90, MT=1 tables have been drawn up using a special program and have exactly the same format, except that the “E-GAM”, “E-BET” and “E-ALF” columns contain the total energy releases in the whole radioactive decay chain right up to decay into a stable nucleus or a nucleus with a half-life of more than 3 years.

The tables give the corresponding branching ratios (in fractions and not in percentages) for those nuclei whereby, during decay, daughter nuclei may form not only in the ground state but also in the metastable states.

The tables also give data for stable isotopes: the abundance of each isotope in its natural mixture. The library of basic decay data now contains information on 2361 nuclides. By way of example, Table 1 gives decay data for Po.

2. Radioactive decay photon spectra

Multiplicities and photon energy spectra emitted during radioactive decay are given in an MF=91 table in a 15-group representation from 0 to 11 MeV. The main data sources for this library were the International Commission on Radiological Protection handbook, Ref. [3], and the JEF-2 library. As in the case of the basic decay data, decay photon spectra for all isotopes were collated in one consolidated MF=91 table named after the chemical symbol of the element to which these isotopes relate. Photon spectra are not necessarily given for all the isotopes for which decay data are given. The decay data library has a sign which indicates whether photon spectra data exist for a given radioactive nucleus. At present, the photon spectra library contains data for 819 radionuclides.

Apart from the 15-group photon spectra, the table gives the energies and yields of several (up to four) of the strongest gamma lines. It should be pointed out that the contributions of these lines to the 15-group spectra given were also taken into account. If the lines identified are considered separately in calculating radioactive decay photon propagation in a medium, their contribution should be subtracted from the 15-group spectra given.

Table 1. Decay data for Po

```
NAM=PO        BIB=FOND  MF= 90  MT=  0   Z=84.
              LT =  26  LC= 12  LS= 12  LF= (I4,a2,e6.0,a1,5e7.0,3e8.0)
* A   T1/2     BETA+  BETA-    IT   ALFA    SF    E-GAM   E-LOC   E-ALF
2020 -  44.7m  98.0                  2.0          8.40-1  1.58-1  1.11-1
2030 -  36.7m  99.89                 0.11         1.63+0  1.47-1  5.90-3
2031 -   1.2m   4.5          95.5                 1.60+0  2.36-1
2040 -  3.53h  99.34                 0.66         1.15+0  1.50-1  3.55-2
2050 -   1.8h  99.96                 0.04         1.59+0  5.32-2  2.10-3
2060 B   8.8d  94.55                5.45          1.19+0  1.41-1  2.84-1
     M                                0.0
2070 -  350.m  99.97                 0.03         1.29+0  4.22-2  1.10-3
2071 -   2.8s                100.                 1.088   2.88-1
2080 B 2.898y  2.2-3                 100.         1.90-5  1.13-7  5.115
     M                                0.0
2090 -  102.y   0.26               99.74          5.15-3  4.40-4  4.97+0
2100 - 138.4d                        100.         8.50-6  8.18-8  5.40+0
2110 B 0.516s                        100.         7.70-3  2.00-4  7.59+0
     M                                1.0
2111 B  25.2s                        100.         1.49+0  1.00-2  7.55+0
     M                                1.0
2120 -  3.-7s                        100.         0.0      0.0     8.95+0
2121  1.42-8s                  87.    13.         1.12+0  1.23-1  1.34+0
2122 -  45.1s                        100.         9.10-2  3.70-4  8.95+0
2130 - 4.2-6s                        100.         0.0      0.0     8.54+0
2140 -1.64-4s                        100.         8.33-5  8.19-7  7.83+0
2150 -1.78-3s             .04       99.96         1.76-4  6.30-6  7.52+0
2160 -  0.15s                        100.         1.69-5  1.61-7  6.91+0
2170 -    9.s              5.         95.         0.      0.      6.329
2180 -  3.05m            0.02       99.98         9.12-6   1.42-5  6.11+0
*
```

When the 15-group spectra were being plotted, the yields were corrected in such a way as to conserve in the group approximation the total energy of the photons in each energy group. Since the same (average group) energy is ascribed in this approximation to all photons in a given group, the group multiplicity for energy conservation is calculated according to the formula:

$$\Lambda_j = \sum_n \lambda_n \times E_{\gamma,n} / \bar{E}_j + \int_{\Delta E_j} \lambda(E_\gamma) \times (E_\gamma / \bar{E}_j) dE_\gamma \tag{1}$$

Here λ_n and $E_{\gamma n}$ are the yield and energy of the n-th line in the energy group under consideration, $\lambda(E\gamma)$ is the continuous spectrum of the multiplicity of decay photons, and $\bar{E}_j$ is the average energy of the photons in the group.

The format of the MF=99 tables, containing data on radioactive decay photon spectra, is described in Ref. [5] and illustrated in Table 2 by the example of tungsten isotopes

Table 2. Photon spectra emitted during the decay of W isotopes

```
NAM=W        BIB=BN93  MF= 99  MT=  0   Z = 74.         DAT=0312.92
LV = 10      LT =  24  LC=  8  LS=  8  LF = (I4,9E7.0)
*            Radioactive decay photon yields.
*               15 groups and the 4 strongest lines
       1769   1779   1789   1799   1819   1859   1879   1889
*
   1
   2
   3
   4
   5
   6
   7
   8         .0109
   9         .3575                               .0386
  10         .4667                               .7249
  11         .2536                .0004          .0     .0068
  12  .7174  .6026         .0017  .0    2.32-4  .1104  .0
  13 1.5050 1.5750  .2117  .7617  .6417 3.38-4  .4167  .0033
  14  .0     .0     .0     .3667  .0    0.0     .0     .0
  15  .2933  .2867  .1133  .2000  .1200 8.39-5  .2200  .0
*
 101        1.0360                      2.13-4  .6858
 201         .1020                      0.1254  .2930
 102         .1156                              .4795
 202         .4980                              .2340
 103         .4270
 203         .1310
 104
 204
*
```

Specialized programs were written to verify the compiled libraries. The first of these was for all the nuclides for which a photon spectrum is given; the energy removed by the photons was calculated on the basis of the group yields given. This value was then compared with the value given in the EGAM column of the decay data library. The causes of discrepancies were established and eliminated. The second program was written with the aim of verifying data for the strong lines that had been identified. The verification procedure consisted of determining the energy remaining once the contribution from the strong lines to each energy group had been subtracted. The remaining value should be zero or positive.

Apart from the aforementioned verification programs, the SUBGAM program was written to calculate the activity and total photon spectrum for a sample of a given composition. The input file includes information on the nuclide composition (nuclear densities) and the user-defined group representation (boundaries of the photon groups). The first calculation is activity, and beta and gamma energy release in the nuclides, using data on the half-lives and energy release on decay from the decay data library. The full photon spectrum is then calculated for a given sample, either in the basic 15-group representation or in another, user-defined, grouping. The SUBGAM program was used to create the MF=91, MT=1 tables which contain data on the spectra of all the photons in the decay chain. SUBGAM is now sometimes used as an independent program.

3. Fission product yields

At the start of the 1970s several evaluations of fission product yields appeared which had been carried out in various laboratories around the world for the basic fissile nuclei. The results of the fission product yield evaluations differed because of the lack of available experimental data. The most comprehensive experimental information available is for thermal neutron fission of ^{235}U, although even in this case it is not possible to measure the independent yields of many short-lived products. There is less experimental information for thermal neutron fission of ^{239}Pu and “fast” neutron fission of ^{235}U (i.e. induced by neutrons with a spectrum close to the part of the fission neutron spectrum that is higher than 1 MeV), although the quality is the same as for ^{235}U. Various semi-empirical methods are being used to evaluate the unmeasured low yields for these nuclei. Of course, there is at times extensive scatter of the evaluated data for these kinds of fission products but, in practical terms, low yield products (which are also, as a rule, very short-lived) do not have a high value and these discrepancies are not significant.

The situation regarding the energy dependence of fission product yields still leaves something to be desired. Even for basic fuel nuclides, detailed energy dependences of yields have not been measured - not even for maximum yield products. The experimental data obtained are for thermal neutrons, “fast” neutrons or neutrons with an energy of the order of 14 MeV - (D-T) neutrons. Here, the “fast” neutron spectrum is usually not particularly well known: it is a fast reactor spectrum, similar to a fission spectrum only in its “hard” part. Accordingly, evaluated data are also given at three “points”: for thermal, fast and 14 MeV neutrons. The user may of course wonder how such scant data on yield energy dependence are to be used, but the evaluators have no clear answer.

There are considerably fewer data on the fission product yields for thorium and ^{233}U, and the available evaluations for minor actinides are based only on semi-empirical methods. The data in these cases are clearly not highly reliable.

The main evaluations of recent years are included in the ENDF/B-6, JENDL3 and JEF2 libraries. We had access only to data from the ENDF/B-6 and JENDL-3.2 libraries. The contents of these libraries is given in Table 3.

Table 3. Content of the ENDF/B-6 and JENDL-3 libraries with respect to independent fission product yields

Actinide	**ENDF/B-6**					**JENDL-3**				
	MAT	N	T	F	H	MAT	N	T	F	H
90-Th-227	9025	1175	+							
90-Th-229	9031	1196	+							
90-Th-232	9040	1300,1303		+	+	3905	1230		+	+
91-Pa-231	9131	1223		+						
92-U-232	9219	1195	+							
92-U-233	9222	1223.1215,1207	+	+	+	3922	1230	+	+	+
92-U-234	9225	1228.1211		+	+					
92-U-235	9228	1248.1252,1226	+	+	+	3924	1230	+	+	+
92-U-236	9231	1275.1243		+	+	3925	1230	+		
92-U-237	9234	1286		+						
92-U-238	9237	1303.1276		+	+	3926	1230		+	+
93-Np-237	9347	1239.1232		+	+	3931	1230		+	
93-Np-238	9349	1260		+						
94-Pu-238	9434	1222		+						
94-Pu-239	9437	1228.1224,1204	+	+	+	3943	1230	+	+	+
94-Pu-240	9440	1236.1219		+	+					
94-Pu-241	9443	1258.1257	+	+						
94-Pu-242	9446	1265		+		3946	1230	+		
95-Am-241	9543	1218.1215,1198	+	+	+	3951	61	+		
95-Am-242m	9547	1231	+							
95-Am-243	9549	1231		+		3954	61	+		
96-Cm-242	9631	1186		+						
96-Cm-245	9640	1236	+							
98-Cm-249	9852	1207	+							
98-Cm-251	9858	1237	+							

The ENDF/B-6 library contains the evaluation by England and Rider (T.R. England and B.F. Rider, 1992). It includes 60 sets of independent and correspondingly cumulative fission product yields for 36 fissile nuclei. These yields are given for one or more fission neutron energies per nucleus, and in some cases data are also given for spontaneous fission. The number of products in each set varies, but the average number of fission products is roughly 1200. This evaluation included more than 3000 new experimental values.

The JENDL-3 library includes the earlier evaluation by Rider and Meek (B.F. Rider and M.F. Meek, 1989). Although it has much fewer evaluated sets compared with ENDF/B-6, it does have data on all the main actinides. The yields for some fissile nuclei are given in the energy dependence: for thermal neutrons (0.0253 eV), fast neutrons and neutrons with an energy of 14 MeV. The yield sets for all the nuclei contain the same number of fission products - 1230.

Both libraries use the ENDF/B format, and they are not used to their full extent in calculation programs either in the Russian Federation or abroad. It is common practice to establish program sublibraries of fission product yields on the basis of these data or any other evaluations for use in solving specific tasks and with their own program formats. Therefore, when calculating, analyzing and comparing the results obtained from the various programs, the differences in the fission product yields in the libraries used must always be identified and taken into account. An attempt to overcome this problem was the decision to establish a library of independent fission product yields in the constant set for a wide number of users, i.e. with general purpose comprehensive data.

Analysis of the data from the ENDF/B-6 and JENDL-3.2 libraries showed that there was no reason to give preference to either. It was decided to adopt the data from the ENDF/B-6 library as a basis, since they were more comprehensive, but to standardize the number of products in each set, i.e. 1234. This number includes all the fission products for which data is also given in JENDL-3.2 and several isomers indicated only in ENDF/B-6.

Differences in the fission product sets, for which data are given for various actinides in these libraries, are attributable to products with very small yields and lifetimes. Products not included in the ABBN-93 set, but which figure in the ENDF/B-6 data, were discarded and their yields added to the independent yields of the nearest members of the isomeric chains. Where the nuclide set included in ENDF/B-6 was smaller than that adopted in ABBN-93, the set was supplemented on the basis of data from JENDL-3.2, so that the cumulative yield of the first member of the isomeric chain adopted in ENDF/B-6 (but not the first for the longer chain adopted in ABBN-93) remained the same as in the original ENDF/B-6 library. As regards the fission product radioactive decay schemes, they are practically the same in ABBN-93 as in JENDL-3.2 except for minor corrections made to take the formation of the newly introduced isomers into account, and corrections to several errors detected in the branching ratios.

Fission product yields - $y_{i,p}^{d}$ - are given for all the main fuel nuclides, i, for all the significant fission products, p, in the three energy ranges, d. Independent fission product yields are given in the ABBN MF=72 table. The first energy range comprises groups 26 to 8. In this range it is recommended to use the yield values measured at the thermal energy of the neutrons inducing fission. Exceptions are fuel nuclides which are non-fissile or not easily fissile when exposed to thermal neutrons. For these nuclides in the first energy range, it is recommended to use the yields measured on the fission spectrum or spectra close to it. Data for the first range are given in the table with MT=1.

The second range comprises the groups 7 to 2 and yields are given only if they differ from those given for the first range. Data for the second range are given in the table with MT=2.

The third range comprises groups 1. 0 and -1 (i.e. the region above the (n,n'f) reaction threshold). In this range it is recommended to use the yields measured at an energy of 14-15 MeV. If yields in the third range are given for any nuclide, then yields in the first range are intentionally given.

Data for the third range are given in the table with MT=3. The following default rules are recommended:

- if there are no data for the second range for a given nuclide, data for the first range should be used;

- if there are no data for the third range, data for the second range should be used; and

- if these data are not available either, data for the first range should be used.

In thermal neutron reactor calculations, fission product yields may be calculated on the basis of data only for the first range: the correction for the difference in yields in the second range in that event is, at most, a few per cent for the yield values even in the minimum region when A=110-120. In the same way (and with even greater justification), data only for the first two ranges can be used in fast neutron reactor calculations. Independent yields are normalized to 2.

Data for the ranges are presented in three files called yield1.tab, yield2.tab and yield3.tab respectively for the fission-inducing neutron energy. Table 4 is a fragment from the start of a file showing thermal neutron yields.

Table 4. Fragment of fission product yield

NAM=FP		BIB=END6	MF= 72	MT= 1	DAT= .95	
LV	=10	LT =1235	LC= 20	LS= 7	LF = (I6,6E11.0)	
*	Independent fission product yields normalized to 2.					
*IDENT	lamda	TH-227	TH-229	U-232	U-233	U-235
902270	902290	922320	922330	922350		
* A=66						
230660	1.8734+02	1.3182-18	1.2852-18	1.0474-14	4.1407-19	2.1450-19
240660	2.5941+00	2.5799-14	4.9596-14	2.0499-10	1.6100-14	2.1500-14
250660	3.9496+00	1.1100-11	6.1397-12	2.1100-7	8.7300-12	7.5102-12
260660	1.8908-01	2.6900-10	1.7499-10	1.1620-5	6.7600-10	3.3401-10
270660	2.4889-01	4.9000-10	1.0100-10	4.3990-5	1.2700- 9	2.9301-10
280660	3.5071-06	7.4000-11	1.8299-11	1.4010- 5	5.3900-10	5.1501-11
290660	2.2652-03	7.2100-13	5.1897-14	3.1400- 7	5.3900-12	1.4900-13
300660	0.0000+0	0.0000+0	0.0000+0	3.8100-10	0.0000+0	0.0000+0
* A=67						
240670	6.1449+00	4.0744-15	5.0204-15	7.6896-12	6.0662-15	4.0488-15

The vast majority of practical applications require calculation of the concentrations of the fission products contained in fuel after the fairly considerable time needed to remove the fuel from the reactor core once irradiation is over. Thus, in calculating fission product concentrations it is expedient to start the chains from the isobar with the maximum Z, for which the decay constant, λ, does not exceed the given λ_{max}. This isobar becomes, consequently, the "lowest order" member of the chain. It is assumed that before the start of the calculation the program accessing the ABBN tables for data on decay schemes and fission product yields, using the given λ_{max}, reduces the isobar chains and the independent yield tables, leaving in them only those fission products which are essential for resolving the task (for example, those that may have an influence on the gamma activity of the irradiated fuel after its removal from the reactor core). At the same time, the independent yields of the "lowest order" members of the isobar chains should be replaced with the cumulative yields calculated on the basis of the complete decay schemes given in the MF=71 table.

The data from these three libraries have so far been used in an international test, Ref. [6] to determine the radioactive properties of fresh and spent uranium and MOX fuel. The library of yields has been used for theoretical analysis of the composition of actinide samples after long-term irradiation in the core of a BN-350 reactor, Ref. [7]. All calculations were carried out with the help of the SKALA software package, Ref. [8], using the ABBN-93 libraries as the constant basis, including the decay data and fission product yield libraries described above.

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NOTES ON THE WIMS/ABBN CODE

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A standard description of the WIMS/ABBN code is presented. The code may be requested from the Minatom Code Fund via the Internet http://ultra.ippe.obninsk.ru:8097).

Introduction

A description of the WIMS/ABBN program package is provided, compiled in accordance with a recommendation by the Sectoral Fund for Algorithms and Programs for Nuclear Reactor Calculations and intended to inform potential users of this software product, which is to be transferred to the Sectoral Fund.

The code is designed for engineering calculations of heterogeneous thermal-neutron reactor cells and clusters of several different cells in one- or two-dimensional geometry, in a group approximation using first collision probability methods or the discrete ordinates method. The calculations can take into consideration fuel and absorber burnup, change in the nuclide composition of the fuel, and accumulation of fission products. It is possible to calculate homogenized micro- and macroconstants, neutron reaction rate distributions, neutron lifetimes etc.

Calculations may be performed taking into account burnup and with calculation of fission product accumulation on the basis of ABBN-69gr data, and certain characteristics of delayed neutrons may also be calculated.

Homogenized constants are generated for subsequent use in reactor calculations in the standard WIMS/D4 output format or the ANISN format.

The major difference between the WIMS/ABBN package and the well-known WIMS/D4 program on which it is based is that the previous obsolete system of 69-group constants is replaced by a new system obtained, like the ABBN-93 constants system, on the basis of the FOND-2.2 evaluated neutron data file library. This replacement made it necessary to update the program to take into account a larger number of resonance isotopes, a larger number of channels for actinide transmutation through neutron reactions ((n,2n) reactions and branching of the process as a result of americium-242 isotope formation are taken into account), etc.

1. Name of the code

WIMS/ABBN - program package for calculating heterogeneous reactor lattice cells including burnup and also the accumulation of fission products, with homogenized micro and macro few-group constants being obtained.

2. Computer for which the program is designed

IBM PC.

3. Problem to be solved

The main feature of the code is the use of a new constant base - a 69-group constants system obtained, like the ABBN-93 universal constants system [1], on the basis of the FOND-2.2 evaluated neutron data library [2].

The code is designed for the calculation of neutron fields in cells or clusters of several different cells of a heterogeneous thermal-neutron reactor lattice and for the preparation of homogenized micro and macro few-group constants, taking into account burnup and accumulation of actinides and fission products. The package has a central unit designed to launch the constituent programs and to enable them to interact with the module that carries out the preliminary preparation of the WIMS/ABBN library for calculation. The package has five external data libraries: the WIMS/ABBN library - the main 69-group library of constants in the standard WIMS/D4 format, the ABBN-69gr library for the program that calculates one-group constants of secondary nuclides and fission products, a library of self-shielding factors for plutonium-242 cross-sections in the thermal energy range, a library of fission spectra for various types of fuel, and a library of first moments of moderator scattering for the program that records macroconstants in the ANISN format. Operation of the central unit is controlled by the code's configuration file, where all necessary data and libraries are indicated. The output data have the standard format of ABBN text libraries. The code can present a balance sheet of reaction rates by isotope. The program prints the output data and also displays useful on-screen information regarding the status of the code.

Depending on the task facing the user, the following data may be calculated: few-group homogenized macro and self-shielded sufficient micro cross-sections for the isotope; neutron absorption and formation balance with breakdown by nuclides; non-self-shielded cross-sections for secondary actinides and fission products; delayed neutron lifetime; fission neutron importance; concentrations of isotopes by burnup steps, and also all other characteristics that can be calculated by the standard version of WIMS/D4.

Homogenized constants are generated for subsequent use in reactor calculations in the standard WIMS/D4 output format or the ANISN format.

4. Solution method

In order to solve the transfer equation and calculate the neutron spectrum, standard WIMS/D4 methods are used: the first collision probability method and the discrete ordinates method. The constants are prepared taking into account resonance self-shielding factors.

The homogenized constants are calculated on the basis of reaction rates calculated by WIMS through convolution by spatial zones with weighting for volumes and group neutron fluxes.

The one-group constants for fission products are calculated on the basis of the 69-group spectrum obtained in the basic transport calculation through convolution by groups of microconstants from the ABBN-69gr library. For isotopes present in the WIMS transport calculation, one-group constants are prepared on the basis of constants calculated by WIMS taking into account self-shielding.

Data on delayed neutrons are calculated by burnup steps by solving an adjoint problem based on the 69-group macroconstants and spectra obtained by WIMS.

The WIMS/ABBN library (129 nuclides) and the ABBN-69gr library (106 nuclides) are obtained on the basis of the FOND-2.2 evaluated neutron data libraries in the same models and approximations as the ABBN-93 library.

The calculation of the change in the nuclide composition of fuel as burnup proceeds, takes into account (n,2n) reactions in the main fuel nuclides and branching of the transmutation process as a result of formation of the ground and isomeric states of americium-242. Both the burnup of Eu, Gd, Dy, Er and Hf absorbers and the burnup of fission products with a high absorption cross-section - isotopes and isomers of Pm, Sm, Eu, Ru, Rh, Pd, Xe and Cs - can be taken into account.

5. Limitations on the complexity of the problem

- The number of energy groups of the homogenized constants is ≤69.
- The number of isotopes involved in the calculation and having resonance tables is ≤10.
- The number of WIMS isotopes is ≤129.
- The number of ABBN isotopes for obtaining homogenized constants maxFil is ≤200.
- The maximum number of burnup steps for obtaining homogenized constants maxCycle is ≤200.
- The maximum number of zones for obtaining homogenized constants maxZon is ≤40.

6. Time required to solve a typical problem

The time is determined by the type of calculation, the number of zones in which constants have to be convoluted, the number of burnup steps etc. Six seconds are required to calculate a WWER-1000 reactor cell in a three-zone geometry in 69 groups with output of few-group constants on a Pentium-200. Correspondingly, if a calculation including burnup is required, the time increases by a factor of k, where k is the number of burnup steps.

Calculating one-group constants for 106 isotopes from the ABBN-69 library and 55 isotopes from the main WIMS/ABBN library for one burnup step takes 2.5 seconds.

7. Features of the program

The WIMS/ABBN package is designed as a set of programs. The head program WIMS.EXE calls up the WIMS/ABBN cell calculation program, the AVERAGE.EXE program for calculating one-group constants, and also the LBUILD.EXE program for sorting and preparing the constants library in the ABBN format. The following programs can be run independently of the whole package:

ADJOIN.EXE	-	calculation of certain characteristics of delayed neutrons;
TOANISN.EXE	-	recording of macro cross-sections in the ANISN format;
NORMFLX.EXE	-	presentation of k_0, k_{inf}, k_{eff}, B^2 by burnup steps;
REACTION.EXE	-	presentation of the reaction rate balance sheet;
DENSITY.EXE	-	presentation of concentrations by burnup steps;
FLUXGRF.EXE	-	processing of the spectrum obtained in the transport calculation;
FUNCT.EXE	-	calculation of spectral indices.

The code is controlled by a single WIMS.CFG configuration file which indicates the programs to be used, defines input and output files and libraries, and indicates operating modes. The file is divided into sections each of which has its own name. The main section, which does not have a name, controls the operation of the WIMSABBN.EXE cell calculation program. The remaining programs in the package read their own and the main section. The programs in the package are independent and can be run separately from it.

By comparison with the standard WIMS/D4, the capabilities of WIMS/ABBN have been substantially expanded. The changes affect primarily the WIMS/ABBN neutron constants library - the library has been completely replaced and expanded. Secondly, the list of reactions taken into account in the calculation of the isotope kinetics has been expanded: a number of (n,2n) reactions and the ^{241}Am(n,γ)^{242m}Am reaction have been included.

Note that in order to introduce these new capabilities, no change was required in the WIMS/D4 library format. WIMS/ABBN is fully operational with the standard WIMS/D4 library.

The WIMSABBN.EXE program includes a code which outputs constants and other data into external files in various formats. Above all, this is 69-FLUX, a special file in which the most complete data are entered: macro- and microconstants, spectra, concentrations, multiplication coefficients, bucklings, zone volumes, normalization factors, burnup parameters, etc. The data are outputted by burnup steps. 69-Flux is the standard text file for interaction between the programs forming part of the WIMS/ABBN package. It can also be used separately. Another output file is ABBN/EXCHANGE, a standard exchange file for interaction with other codes and programs which can be accompanied by the ABBN laboratory. The structure and format of the data in this file are designed for interaction with isotope kinetics calculation programs. Depending on the operating modes, data prepared by WIMS for the WIMSABBN.EXE isotope kinetics calculation unit or data from the few-group constant preparation unit may be taken for this file. WIMS/ABBN will read the ABBN/EXCHANGE file, provided this is foreseen by the task list, i.e. by the WIMS.CFG file option, and will replace the previous nuclear concentrations of the isotopes with those newly entered in this file.

The PUNCH file contains macroconstants prepared by the SCRAMBLE WIMS unit in a format designed for the program for calculating delayed neutron characteristics.

The size of the main WIMS/ABBN array is fixed dynamically. It is fixed in the WIMS.CFG configuration file by the user and may be selected for each task in an optimum manner.

For operational ease, the library and the calculation task list file have a description section, the information from which is always included in the output listing.

Useful information is displayed on screen during operation.

8. Auxiliary and secondary programs

WIMS.EXE	-	central program for controlling the code and adapting the WIMS/ABBN library to the time of calculation, i.e. replacement of the ^{235}U fission spectrum with the spectrum indicated by the user and self-shielding of the first ^{242}Pu resonance in the thermal energy region;
ADJOIN.EXE	-	calculation of certain characteristics of delayed neutrons;
TOANISN.EXE	-	recording of macro cross-sections in the ANISN format;
NORMFLX.EXE	-	presentation of k_0, k_{inf}, k_{eff}, B^2 by burnup steps;
REACTION.EXE	-	calculation of the balance sheet;
DENSITY.EXE	-	presentation of concentrations by burnup steps;
FLUXGRF.EXE	-	program for processing the spectrum from the transport calculation;
FUNCT.EXE	-	calculation of spectral indices.

9. Status of the program

The WIMS/ABBN code is currently being used at the Institute for Physics and Power Engineering in Obninsk.

The operation of all units and programs has been debugged and checked in a large number of tests.

The methods and approximations that can be used for neutron calculation are described in Ref. [3] and also in Refs [4, 5, 6].

The individual programs included in the package are described in Refs [7, 8].

The method for preparing constants is described in Refs [7, 9].

The code verification results are presented in Refs [7, 8].

10. Computer requirements

Standard software (DOS, WINDOWS, WINDOWS NT). The size of the operating memory is determined by the problem. Minimum requirement is 18 MB.

11. Programming language

Core WIMSABBN.EXE - FORTRAN-77.
Secondary programs MS FORTRAN-77, ASSEMBLER.

12. Operating system

DOS, WINDOWS NT.

LAHEY - for IBM PC, MS FORTRAN 5.10.

13. Additional information

For most problems, a main array size of 100 000 is sufficient. In this case, less than 16 MB of memory are required. The whole code including the libraries occupies about 3.1 MB on disk.

14. Authors of program

The standard WIMS/D4 code was corrected and updated to WIMS/ABBN by G.M. Zherdev, who also wrote all the remaining programs in the package. WIMS/ABBN was verified and validated by G.M. Zherdev and A.M. Tsibulya. IPPE, pl. Bondarenko, 1, 249020 Obninsk, Kaluga region.

15. Available materials

The program text, libraries and programs are available on hard disk and diskette. The verification report containing detailed instructions for use is available in electronic form. All materials may be obtained by the usual means through the Sectoral Fund for Algorithms and Programs for Nuclear Reactor Calculations (http://ultra.ippe.obninsk.ru:8097).

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