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" NUCLEAR DATA REVIEW AND COMPILATION FOR ATW SYSTEMS."

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1. Introduction.

In order to solve the problem of nuclear power waste transmutation in neutron flux it is necessary to know the characteristics of neutron interaction for a great number of nuclei in the energy range from 0 to hundreds of MeV. We distinguished the most important aspect of this problem - that one of nuclear data for actinides. (from Th to Cm isotopes). We have given the overview of evaluations of characteristic of interaction between neutrons and these nuclei leading to transformation from target-nucleus to neighboring actinide-nucleus or fission fragments in the limited energy range from 0 to 14 MeV.

The review was carried out by comparison of mentioned characteristics from the modern versions of ENDL-82, JENDL-3, ENDF/B-6 and BROND-2 neutron evaluated data among themselves and with recommended data of previous publications and, in some cases, with the measurement results. ENDL-82 and ENDF/B-6 versions were made in USA laboratories, JENDL-3 was made in the laboratories of Japan and BROND-2 version was made in the laboratories of former USSR. The comparison of nuclear data from various libraries was carried out by the most economic method permitting, nevertheless, fully judge of available uncertainties in the knowledge of competitive nuclear data which are important from the point of view of problem of transmutation in various energies neutron flux. The following characteristics were considered:

- fission and capture cross-sections at thermal point ($E_n = 0.0253$ eV)
- below designated as σ_{f0} and $\sigma_{\gamma 0}$, respectively;

- infinitely dilute resonance integrals (hereafter resonance integrals) of fission and capture designated by I_f and I_γ ;
- averaged on ^{252}Cf spontaneous fission neutron spectrum cross-sections of fission, capture and the (n,2n) reactions, designated as $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$ and $\bar{\sigma}_{2n}$;
- cross-sections of fission and the (n,2n), (n,3n) reactions at the point $E_n = 14 \text{ MeV}$, designated as σ_f , σ_{2n} and σ_{3n} .
- fission and capture resonance integrals for a interval of sets with the increasing upper ($E_{\max}$) and lower ($E_{\min}$) limits of integral denoted as ΔI_f and ΔI_γ ;
- averaged cross-sections

$$\langle \sigma_\gamma \rangle = \frac{\int_{E_{\min}}^{E_{\max}} \sigma_\gamma(E) \cdot dE}{(E_{\max} - E_{\min})}$$

for the processes of fission ($\gamma=f$) and capture and number of sets of integration limits ($E_{\max}$, $E_{\min}$).

Before the comparison of various data let us note some comments. Experimental cross-section values at thermal point can be obtained by some various methods:

1. the measurements by time-of-flight method for neutron energy 0.0253 eV;
2. the measurements on Maxwell spectra with the average energy 0.0253 eV in thermal columns;
3. the measurements on reactor spectra having the average energy 0.0253eV.

The experimental results of the first group have priority over the rest ones. The discrepancies between the data of these groups are

observed in practice, and the data of the third group are met more often. As an illustration there is a Table from Ref.[1] where the comparison of cross-sections at thermal point is given for a set of actinides with the use of the first and second methods.

Table 1

The values of σ_{fo} and $\sigma_{\gamma o}$, in barn

nucleus	σ_{fo}		$\sigma_{\gamma o}$	
	method 2	method 3	method 2	method 3
$^{234}_{U}$	0	0	94.7	84.1
$^{236}_{U}$	0	0	8.54	7.66
$^{237}_{Np}$	-	-	151	161
$^{238}_{Pu}$	17.5	-	500	434
$^{240}_{Pu}$	0.06	0.05	246	235
$^{241}_{Pu}$	950	1030	345	390
$^{242}_{Pu}$	-	-	15.7	14.3
$^{241}_{Am}$	2.91	3.49	622	800
$^{243}_{Am}$	0	0	142	147
$^{244}_{Cm}$	0	0	9	8

According to the method of data obtaining experimentally evaluated resonance integrals may be classified into the following most common groups :

1. calculation on measured differential data;
2. measured by activation method with Cd-shield for relatively "clean" samples data;
3. measured on the reactor by various ways data;

The list of evaluation methods is discussed in more detail in [2] and the data of the first and second groups have priority in this case.

The values of I_f and I_γ depended on the choice of $E_{\min}$ and $E_{\max}$. Dependence on $E_{\min}$ is important both for I_f and I_γ under strong resonance distinctions in energy dependencies of corresponding cross-sections. In the experiments $E_{\min}$ changes from 0.3 to 0.68 eV. The value of $E_{\min}=0.5$ eV is the mostly used value and it will be taken by us for our further estimations. The choice of $E_{\max}$ is most important for fission reaction, particularly, in case of the threshold structure of this reaction. As it was noted in [3] the calculations for ^{243}Am has shown that the change of $E_{\max} = 10$ MeV by $E_{\max} = 20$ MeV increases the I_f value a 1.5 times. However at comparison of results of experimental evaluation of resonance integrals obtained either on thermal columns, or immediately in reactors, one should remember that neutrons with the energy $E_n > 10$ MeV are practically absent in the reactors. Therefore we have restricted ourselves to the energy $E_{\max}=10$ MeV. It must be noted also, that I_f values obtained at the reactors may contain an error that was caused by the differences of reactor spectra at $E_n > 100$ keV from Fermi ones and was not taken into account.

The calculation of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$ and $\bar{\sigma}_{2n}$ was carried out with the help of approximation of spectrum of ^{252}Cf spontaneous fission neutrons from [4], that was recommended by US National Bureau of Standard as a standard reference.

At least, the comparison of ΔI_f , ΔI_γ , $\langle \sigma_f \rangle$ and $\langle \sigma_\gamma \rangle$ values for various $(E_{\min}, E_{\max})$ pairs was carried out for visual identification of energy intervals where the differences of

competitive versions of evaluated data are of a great significance. RECONR code was used under the recovery of energy dependence of cross-sections in resonance region. The accuracy of recovery corresponded to the $ERR = 0.05$ value of parameter in all cases, except ^{232}Th and ^{238}U . In case of ^{232}Th we were forced to restrict ourselves by the value $ERR = 0.1$ and for ^{238}U this limit was equal to 0.5. Thereat the I_f and I_γ calculation error was less than that of recommended values of these quantities.

2. The isotopes of $^{232-233}\text{Th}$.

2.1 Th-232.

Even-even ^{232}Th nuclei is a long-lived one with a period $T_{1/2} = 1.405 \cdot 10^{10}$ years (α - decay). Its abundance consists for 100%. ^{232}Th nucleus is a primary raw material for ^{233}U generation in neutron flux by the $^{232}\text{Th} (n, \gamma) ^{233}\text{Th} (\beta^-) ^{233}\text{Pa} (\beta^-) ^{233}\text{U}$ reaction chain. One can ignore ^{232}Th partial recovery in the $^{233}\text{Th} (n, 2n)$ reaction because of extremely small $T_{1/2} = 22.3$ min value for $^{233}\text{Th} \beta^-$ - decay. The $^{232}\text{Th} (n, \gamma)$ reaction run is most intensive in resonance neutron flux. ^{232}Th nuclei is fissionable by thermal neutrons with a very small probability. Only in the range of neutron fission the process of ^{232}Th nucleus fission by neutrons begins to concurrent with radiation capture process. The $^{232}\text{Th} (n, 2n)$ reaction becomes predominate at the region of thermonuclear neutrons ($E_n = 14$ MeV).

The characteristics of interaction of neutrons with ^{232}Th nuclei are well investigated both in the experiments and in theory, and are represented in all versions of modern evaluated neutron data compared by us.

R. Howerton (LNLL, USA) is the author of ENDL-2 (MAT 7864) evaluation; 1983 is the year of evaluation completion.

BROND-2 (MAT 9032) evaluation is based on original data of [Wi66] modified to ENDF/B-4 format by O.Ozer in 1974. In [Ma84] the new evaluation in resonance region was carried out and BROND-2 file was compiled. A resonance region was formed for two pseudo-isotopes. S- and d- resonances only were accounted for the first isotope and p-resonance was accounted for the second one. The determination of resonance parameters was carried out from the results of statistical

analysis from [Mu81] with account of missed levels. The region of resolved s-resonances is between 5 and 3900 eV and contains 240 levels. For p-resonances the same region is between 1 eV and 1000 eV. The region of unresolved resonances is between 1 keV and 200 keV. Cross-sections there are described by energy dependent widths. The strength functions were obtained on the results of an analysis carried out by the authors of [Ma83] according to the data of capture and inelastic scattering cross-section measurements. All cross-sections in the range from 10^{-5} eV to 5 eV were obtained without any additions from contributions of resonance dependencies of the range with $E_n > 5$ eV. In the energy range from 5 eV to 1 keV the cross-sections are completely determined by resonance parameters with background (nonresonance) cross-section which is equal to zero. In the range from 1 keV to 200 keV nonzero background cross-section is due to effective account of f - resonances. In the range from 200 keV to 1 MeV cross-sections are based on new evaluation carried out in [Ma83].

M.Bhat, A.Smith, B.Leonard, G.De Saussure and et al. are the authors of ENDF/B-6 evaluation. 1977 is the year of evaluation completion. ENDF/B-6 version differs from ENDF/B-5 with format of data representation only. The evaluation of resonance parameters in the range of resolved resonances is carried out by D. Olsen [37] by averaging of available data. The range of resolved resonances extends to 4 keV. Capture resonance integral over the range from 5 eV to 4 keV is equal to 81.58 barn.

The range of unresolved resonances is between the limits 4 keV and 50 keV. The evaluations of capture cross-section from 4 keV to 25

keV were carried out by G. De Saussure and R. Macklin. Smooth transition to Poenitz evaluation (from 40 keV to 50 keV) takes place above 25 keV. These data were used to obtain the parameters of unresolved resonances by code UR [Pe73]. In the thermal range (10^{-5} eV - 5 eV) capture cross-section evaluations from [Ne77] are renormalized on the value $\sigma_{\gamma 0} = 7.4$ barn and are used in the interval from 10^{-5} eV to $2.53 \cdot 10^{-2}$ eV. The evaluation from [Ch78] is used above $2.53 \cdot 10^{-2}$ eV. The evaluation of capture cross-section above 50 keV was carried out by W. Poenitz according to the ^{235}U fission cross-section evaluation of the same author ([Po78]). He also carried out the ^{232}Th fission cross-section evaluation with the same normalizing. The (n,2n) and (n,3n) reaction cross-section evaluation was carried out by J. Meadows et. al [1]. Experimental data renormalized to a common set standards were used.

T. Oshawa (Kinki Univ., Japan) is the author of JENDL-3 (MAT 3905) evaluation; 1987 is a year of evaluation completion. The range of resolved resonances is between the limits 10^{-5} eV and 3.5 keV. Multi level Breit-Wigner formula is used for cross-section calculations. There are mainly used JENDL-2 resonance parameters modified partly:

- new resonance parameters from [Ko86] were entered for 22 low energy resonances giving the main contribution to resonance integrals;
- average radiation width of 24.7 meV was entered for resonances with unknown radiation widths. The range of unresolved resonances is between the limits 3.5 keV and 50 keV. Average resonance parameters are given. Energy dependencies of S_0 and S_1 strength functions were calculated so as to reproduce the total and capture cross-sections in this region. Below 3.5 keV background cross-section is given for

capture process only. Above 50 keV fission cross-section was determined by the ratio data $\sigma_f(^{232}\text{Th})/\sigma_f(^{235}\text{U})$ measured in [Be77] with the help of $\sigma_f(^{235}\text{U})$ evaluation from [Ma79]. Capture cross-section evaluation is based on the experimental data of [Ko78] and calculations with the code CASTHY [Ig75]. The (n,2n) and (n,3n) reaction cross-sections have been calculated with the model of [Se78].

In Tables 2.1-2.7 the comparison of some characteristics of interaction between neutrons and ^{232}Th from ENDL-82, JENDL-3, ENDF/B-6 and BROND-2 among themselves and with recommended data is given.

Table 2.1

The values of σ_{f0} and $\sigma_{\gamma 0}$ for ^{232}Th , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
σ_{f0}	0	0	0	0	$(39^{+4}_{-4}) 10^{-6}$	1973 [^] [5]
					$< 2.5 10^{-6}$	1984, [6]
$\sigma_{\gamma 0}$	7.43	7.40	7.40	7.40	$7.40^{+0.0}_{-0.0}$	1973, [5]
					$7.37^{+0.0}_{-0.0}$	1984, [6]

Table 2.2

The values of I_f and I_γ for ^{232}Th , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	0.370	0.365	0.350	0.350	-	
		(0.636) ^{*)}				
I_γ	95.5	84.4	85.4	85.9	85^{+3}_{-3}	1973, [5]
		(84.4) ^{*)}	(86.1) ^{*)}		85^{+3}_{-3}	1984, [6]

*) - $E_{\text{max}} = 20 \text{ MeV}$

Table 2.3.

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{232}Th , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	0.081	0.096	0.025	0.355	1.300	0.650
JENDL-3	0.082	0.083	0.022	0.350	1.181	0.800
ENDF/B-6	0.078	0.090	0.023	0.343	1.430	0.620
BROND-2	0.078	0.091	0.025	0.345	1.560	0.500

Table 2.4

The value of ΔI_f for ^{232}Th , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3n	ENDF/B-6	BROND-2
0.5	10	0	0	0	$7 \cdot 10^{-7}$
10	10^2	0	0	0	$3 \cdot 10^{-4}$
10^2	10^3	0	0	0	$1 \cdot 10^{-4}$
10^3	10^4	0	0	0	$1 \cdot 10^{-5}$
10^4	10^5	0	0	0	0
10^5	10^6	0	$3 \cdot 10^{-4}$	$2 \cdot 10^{-4}$	$2 \cdot 10^{-4}$
10^6	10^7	0.370	0.365	0.350	0.349

Table 2.5

The value of ΔI_γ for ^{232}Th , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	1.590	1.652	1.976	1.964
10 10^2	64.57	59.01	58.27	58.57
10^2 10^3	24.26	19.23	20.23	20.52
10^3 10^4	3.459	2.983	3.373	3.496
10^4 10^5	1.121	1.127	1.045	1.057
10^5 10^6	0.346	0.310	0.398	0.409
10^6 10^7	0.119	0.098	0.105	0.105

Table 2.6.

The value of $\langle \sigma \rangle$ for ^{232}Th , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	0	0	0	$5 \cdot 10^{-7}$
10 10^2	0	0	0	$1 \cdot 10^{-4}$
10^2 10^3	0	0	0	$3 \cdot 10^{-5}$
10^3 10^4	0	0	0	$2 \cdot 10^{-6}$
10^4 10^5	0	0	0	0
10^5 10^6	0	$3 \cdot 10^{-4}$	$2 \cdot 10^{-4}$	$2 \cdot 10^{-4}$
10^6 10^7	0.227	0.221	0.213	0.213

Table 2.7

The value of $\langle \sigma_{\gamma} \rangle$ for ^{232}Th , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	0.282	0.305	0.418	0.407
10 10^2	26.01	23.15	22.96	23.23
10^2 10^3	7.181	5.348	5.946	6.003
10^3 10^4	1.140	1.076	1.150	1.177
10^4 10^5	0.384	0.375	0.377	0.379
10^5 10^6	0.165	0.143	0.157	0.159
10^6 10^7	0.030	0.021	0.023	0.024

2.2 ^{233}Th .

Even-odd ^{233}Th nucleus with a half life $T_{1/2}=22.12$ min transforms into ^{233}Pa one due to β^- decay. ^{233}Th nucleus is generated in the ^{232}Th (n, γ) reaction. Because of a small $T_{1/2}$ (^{233}Th) it may be expected, that the quality of evaluation of characteristics of neutron and ^{233}Th nucleus interaction, which is concurrent to β^- decay, should not appreciably effect on ^{233}U generation rate in reactors. Data about characteristics of neutron and ^{233}Th interaction among the versions of evaluated neutron data we used there, are in ENDL-82 and JENDL-3 only.

R. Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7865)

evaluation; 1983 is the year of evaluation completion.

T. Ohsawa (Kinki Univ., Japan) is the author of JENDL-3 evaluation; 1987 is the year of evaluation completion. Because of the absence of experimental data the resonance parameters were not evaluated. Fission and capture cross-sections at 0.0253 eV were obtained with the help of extrapolation by the $1/V$ law in the range above 200 eV. The $1/V$ dependence plus the constant value was assumed above 20 keV. Calculation of the $(n,2n)$ and $(n,3n)$ reaction cross-sections were carried out using an evaporation model of [Se78]. Fission probability derived from the data of [Ba74] on direct reactions was used in the calculations of fission cross-sections. Capture cross-section was calculated by statistic and optical models with the gamma-ray strength function of 0.00352.

In Tables 2.8-2.12 the comparison of some characteristics of neutrons and ^{233}Th nucleus interaction, obtained with the help of ENDL-82 and JENDL-3 evaluations, among themselves and with various years recommended data is given.

Table 2.8

The values of σ_{f0} and $\sigma_{\gamma0}$ for ^{233}Th , in barn.

Value	ENDL-82	JENDL-3	Recommend. value	Year of public.
σ_{f0}	15.03	15.00	15^{+2}_{-2}	1973 [^] [5]
			15^{+2}_{-2}	1984, [6]
$\sigma_{\gamma0}$	1401	1450	1485	1973, [5]
			1500^{+100}_{-100}	1984, [6]

Table 2.9.

The values of I_f and I_γ for ^{233}Th , in barn.

Value	ENDL-82	JENDL-3	Recommend. value	Year of public.
I_f	83.67	10.88 (11.1) ^{*)}	-	
I_γ	417.8	643.0 (643.0) ^{*)}	400^{+100}_{-100}	1984, [6]

*) - $E_{\text{max}} = 20 \text{ MeV}$

Table 2.10

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{233}Th , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	0.127	0.095	0.096	0.338	0.750	1.120
JENDL-3	0.110	0.084	0.087	0.410	1.160	0.917

Table 2.11

The values of ΔI and ΔI for ^{233}Th , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ΔI_f	ΔI_γ		
	ENDL-82	JENDL-3	ENDL-82	JENDL-3
0.5 10	28.01	6.119	326.7	507.5
10 10^2	38.06	1.721	64.37	99.93
10^2 10^3	13.53	1.012	19.55	29.01
10^3 10^4	2.399	0.792	5.020	5.700
10^4 10^5	1.002	0.656	1.310	1.120
10^5 10^6	0.345	0.364	0.700	0.670
10^6 10^7	0.329	0.219	0.110	0.064

Table 2.12

The values of $\langle \sigma \rangle$, $\langle \sigma \rangle$ for ^{233}Th , in barn.

$E_{\min}$ (eV)	$E_{\max}$ (eV)	$\langle \sigma_f \rangle$		$\langle \sigma_\gamma \rangle$	
		ENDL-82	JENDL-3	ENDL-82	JENDL-3
0.5	10	14.75	1.534	76.87	119.3
10	10^2	15.98	0.661	22.61	35.06
10^2	10^3	3.845	0.413	6.497	9.425
10^3	10^4	0.842	0.336	1.687	1.862
10^4	10^5	0.394	0.230	0.544	0.464
10^5	10^6	0.127	0.162	0.165	0.158
10^6	10^7	0.167	0.105	0.026	0.015

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^{232}Th

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3. Isotopes of $^{231-233}\text{Pa}$.

3.1 ^{231}Pa .

Odd-even ^{231}Pa nucleus is a long-lived one ($T_{1/2}=3.276 \cdot 10^4$ years, α -decay). The $^{232}\text{Th}(n,2n)^{231}\text{Th}(\beta^-)^{231}\text{Pa}$ and $^{232}\text{Th}(n,\gamma)^{233}\text{Th}(\beta^-)^{233}\text{Pa}(n,2n)^{232}\text{Pa}(n,2n)^{231}\text{Pa}$ reaction chains are the generation channel in nuclear reactors. There are three channels where transmutation in neutron flux is possible:

- in capture reactions of thermal and resonance neutrons;
- in fission neutron reactions with possible competition from the side of the capture and (n,2n) reactions;
- by thermonuclear neutrons ($E_n = 14$ MeV) in fission reactions with possible competition with the (n,2n) reaction. The account of ^{231}Pa should lead to some closer definition of ^{233}U generation rate because of taking into consideration the $^{231}\text{Pa}(n,\gamma)^{232}\text{Pa}(\beta^-)^{232}\text{U}(n,\gamma)^{233}\text{U}$ transmutation chain. Because of small accessibility of ^{231}Pa the characteristics of its interactions with neutrons are not investigated well enough. In particular, fission process is studied not fully: there are not enough data on cross-section and only a publication about I_f value. Capture cross-sections were not measured at all. The choice of $E_{\min}$ is important for ^{231}Pa in the calculations of I_f and I_γ , since there are two levels: at the energy 0.4 eV (strong resonance) and 0.497 eV (weak resonance). It was established, that fission threshold occurred near 400 keV.

Among the versions of evaluated neutron data we considered, the evaluation of characteristics of neutron and ^{231}Pa nucleus are given in JENDL-3 and ENDF/B-6 only.

T. Ohsawa, M. Inoue, T. Nakagawa (Kinki Univ., Japan) are the

authors of JENDL-3 (MAT 3911) evaluation; 1987 - is the year of evaluation completion. The region of resolved resonances extends to 115 eV. Single-level formalism of Breit-Wigner is used. Neutron and radiative widths were determined mainly according to [Hu81]. Fission widths were estimated by the area under resonances measured in [Pl79]. For the resonance which fission area was not measured, an average value of 40 meV was assumed. A negative resonance was given on the basis of recommendations from [Mu84] to reproduce the proposed thermal cross-section. The region of unresolved resonances was in the limits between 115 eV and 40 keV. Resonance parameters were obtained by averaging of resolved resonance parameters. S_1 value was obtained by optical model calculations. Background cross-section for radiative capture process provides smooth transition to the range of energies above 40 keV. Radiative capture cross-section at these energies was calculated by CASTHY code [Ig75]. Average radiative width and level spacing are normalized on 40 meV and 0.47 eV values obtained from [Hu81], respectively. The (n,2n) and (n,3n) reaction cross-sections were calculated with the model of [Se80]. Fission cross-section at $E < 12$ MeV was evaluated according to measurement results from [Pl79], and at $E > 12$ MeV it was based on the evaluation of [Ma79] adopted after appropriate renormalization.

ENDF/B-6 (MAT 9131) evaluation for ^{231}Pa was preserved the same as ENDF/B-5 (MAT 8131) one, with the exception of format of data representation and characteristics for delayed fission neutrons.

F.Mann (HEDL, USA) is the author of ENDF/B-5 evaluation; 1977 is a year of evaluation completion. Parameters of resolved resonance levels correspond to recommended data from [Mu73] except of fission and radiative resonance widths at negative energy, that are not in a

good agreement with $\sigma_{\gamma 0} = 210 \pm 2$ barn evaluation from the same reference. Average resonance parameters in the range of unresolved resonances (14.3 eV - 1000 eV) were determined by averaging of resolved level parameters. At $E > 1000$ eV capture cross-section was calculated by statistical model according to results from [Ma77]. In the interval from 100 keV to 3 MeV fission cross-section evaluation was obtained according to the results from [Mu71] derived in LASNL under nuclear explosions. At $E < 100$ keV and $E > 3$ MeV fission cross-section was calculated by statistical model from [Ma77]. It was mentioned nothing about the evaluation method of the (n,2n) and (n,3n) reaction cross-sections in description to MAT 9131 from ENDF/B-6.

The comparison of some characteristics of interaction between neutrons and ^{231}Pa nucleus, corresponding to JENDL-3 and ENDF/B-6 evaluations, among themselves and with recommended data is given in Tables 3.1-3.5.

Table 3.1

The values of σ_{f0} and $\sigma_{\gamma0}$ for ^{231}Pa , in barn.

Value	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
σ_{f0}	0.020	0.010	$0.010^{+0.005}_{-0.005}$	1973, [5]
			$0.006^{+0.001}_{-0.001}$	1976, [2]
			$0.020^{+0.001}_{-0.001}$	1984, [6]
			$0.014^{+0.004*}_{-0.004*}$	1993, [3]
			$0.019^{+0.004**}_{-0.004**}$	1993, [3]
			$0.010^{+0.005***}_{-0.005***}$	1993, [3]
$\sigma_{\gamma0}$	200.7	228.0	201^{+22}_{-22}	1974, [2]
			$200.6^{+2.3}_{-2.3}$	1984, [6]
			208^{+19*}_{-19*}	1993, [3]
			201^{+13**}_{-13**}	1993, [3]
			218^{+13***}_{-13***}	1993, [3]

*) - obtained by time-of-flight method cross-section at $E = 0.0253$ eV

**) - averaged over Maxwell spectrum cross-section with the average

energy $\bar{E}_n = 0.0253$ eV

***) - measured over reactor spectrum cross-section with the average

energy $\bar{E}_n = 0.0253$ eV

Table 3.2

The values of I_f and I_γ for ^{231}Pa , in barn.

Value	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
I_f	3.215	3.752	$0.049^{+0.13}_{-0.13}$ *)	1974, [2]
I_γ	625.1	620.7	1500^{+100}_{-100}	1973, [5]
			525^{+60}_{-60}	1984, [6]
			760^{+80}_{-80} **)	1993, [3]
			786^{+15}_{-15}	1993, [3]

*) - this value was measured at $E_{\min} = 0.55$ eV

**) - $E_{\min} = 0.414$ eV

Table 3.3

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{231}Pa , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
JENDL-3	0.842	0.267	0.430	2.080	0.724	0.134
ENDF/B-6	0.997	0.369	0.784	1.972	1.328	0.057

Table 3.4

The values of ΔI_f and ΔI_γ for ^{231}Pa , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ΔI_f		ΔI_γ	
		JENDL-3	ENDF/B-6	JENDL-3	ENDF/B-6
0.5	10	0.039	0.044	378.4	297.3
10	10^2	0.019	0.035	179.4	225.2
10^2	10^3	0.004	0.009	46.77	60.48
10^3	10^4	0.001	0.027	13.60	23.95
10^4	10^5	0.001	0.008	4.750	11.09
10^5	10^6	0.412	0.636	1.880	2.360
10^6	10^7	2.689	2.991	0.260	0.300

Table 3.5

The values of $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{231}Pa , in barn.

$E_{\min}$ $E_{\max}$ (eV)		$\langle \sigma_f \rangle$		$\langle \sigma_\gamma \rangle$	
		JENDL-3	ENDF/B-6	JENDL-3	ENDF/B-6
0.5	10	0.010	0.014	140.8	107.4
10	10^2	0.006	0.011	54.48	73.09
10^2	10^3	0.001	0.003	16.23	21.04
10^3	10^4	0.0006	0.009	4.664	9.321
10^4	10^5	0.0006	0.003	1.924	3.851
10^5	10^6	0.362	0.468	0.520	0.838
10^6	10^7	1.353	1.413	0.055	0.101

3.2 ^{232}Pa .

Odd-odd ^{232}Pa nucleus is a long-lived one with a half-life $T_{1/2}=1.31$ day. It transforms into ^{232}U nucleus due to β -decay. The $^{232}\text{Pa}(\beta^-)^{232}\text{U}(n,\gamma)^{233}\text{U}$ chain is an additional channel for ^{233}U generation in relation to the $^{232}\text{Th}(n,\gamma)^{233}\text{Th}(\beta^-)^{233}\text{Pa}(\beta^-)^{233}\text{U}$ main channel. In neutron flux the following channels may be supplementary ones in ^{232}Pa transforming:

- fission reaction with radiative capture possible competition in thermal neutron range;
- fission and capture reactions in resonance neutron range;
- fission reaction with the (n,2n) reaction possible competition in fission neutron range;

- fission reaction with the (n,2n) reaction competition in thermonuclear neutron range.

^{232}Pa is generated by the $^{231}\text{Pa}(n,\gamma)^{232}\text{Pa}$ channel in thermal and resonance neutron flux and by $^{233}\text{Pa}(n,2n)^{232}\text{Pa}$ channel in thermonuclear one. Because of a small half-life value the characteristics of neutron and ^{232}Pa interaction have not been practically investigated by experimental methods. In particular, the only paper about I_γ measurement with indeterminate $E_{\min}$ value was published and the I_f measurement data are absent at all. :

The values of characteristics of neutron and ^{232}Pa interaction are presented in JENDL-3 only (the ENDF/B-5 version in ENDF/B-6 has not been passed, therefore we don't discuss it here).

N. Takagi (Tokyo Inst. of Technol., Japan) is the author of JENDL-3 evaluation (MAT 3912); 1988 is the year of evaluation completion. The resonance cross-section structure was not evaluated, i.e., the cross-sections are assumed to be "smooth" over the whole energy scale. The (n,2n), (n,3n), (n,4n) reaction cross-sections were calculated by evaporation model. Below 1 eV energy dependence of fission cross-section was assumed by 1/v law with normalizing by recommended in [6] value of 700 b at thermal point. Above 1 eV energy dependence of fission cross-section was, on the average, (i.e., smooth dependence without resonance destinations) assumed to be the same, as for ^{233}U with normalizing by evaluated data from [Be78]. Below 1 eV energy dependence of capture cross-section was assumed to be proportional to 1/v with normalizing by the value 464 b recommended in [6] at thermal point. Capture cross-section shape near 1 eV was adjusted so as to reproduce I_γ recommended value from [6]. Above 1 eV capture cross-section was calculated by CASTHY code [Ig75]. Average

values of radiative widths and level spacing ($\bar{\Gamma}\gamma=0.04$ eV and $\bar{D}=0.417$ eV) were used under determination of γ -strength function.

Calculated characteristics of neutron and ^{232}Pa interaction corresponding to JENDL-3 evaluation are compared among themselves and with available reference data in Tables 3.6-3.9.

Table 3.6

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{232}Pa , in barn.

Value	JENDL-3	Recommend. value	Year of public.
σ_{f0}	700	700^{+100}_{-100}	1973, [5]
		700^{+100}_{-100}	1984, [6]
$\sigma_{\gamma0}$	464	60	1973, [5]
		464^{+95}_{-95}	1976, [2]
		464^{+95}_{-95}	1984, [6]

Table 3.7

The values of I_f , I_γ for ^{232}Pa , in barn.

Value	JENDL-3	Recommend. value	Year of public.
I_f	312.8	-	
	(313.0)		
I_γ	309	298^{+96}_{-96}	1976, [2]
	(309)	300^{+70}_{-70}	1984, [6]

Table 3.8

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{232}Pa , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
JENDL-3	1.095	0.069	0.748	1.381	0.841	0.266

Table 3.9

The values of ΔI_f , ΔI_γ , $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{232}Pa , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ΔI_f	ΔI_γ	$\langle \sigma_f \rangle$	$\langle \sigma_\gamma \rangle$
0.5	10	207.8	184.0	50.78	57.22
10	10^2	62.98	78.62	23.55	29.38
10^2	10^3	24.19	32.61	8.500	11.87
10^3	10^4	8.620	10.28	3.196	3.416
10^4	10^5	4.420	2.490	1.533	1.015
10^5	10^6	2.320	0.960	1.180	0.172
10^6	10^7	2.500	0.040	1.122	0.006

3.3 ^{233}Pa .

Odd-even ^{233}Pa nucleus is an instable one ($T_{1/2}=26.967$ days) transforming according to β^- -decay into ^{233}U nucleus. In nuclear reactors ^{233}Pa may be generated by the $^{232}\text{Th}(n,\gamma)^{233}\text{Th}(\beta^-)^{233}\text{Pa}$ reaction chain and with lesser probability it may be generated by the $^{232}\text{Th}(n,2n)^{231}\text{Th}(\beta^-)^{231}\text{Pa}(n,\gamma)^{232}\text{Pa}(n,\gamma)^{233}\text{Pa}$ reaction chain. The half-life of ^{233}Th is equal to 22.3 min.

Therefore, in order to describe ^{233}U generation in reactors it is important, first, to know the characteristics of neutron absorption by ^{232}Th nucleus, and then to know the same characteristics for ^{233}Pa one with the less rigorous requirements to evaluation accuracy.

The experimental evaluation of these characteristics becomes complicated because of small half-life for ^{233}Pa nucleus. In particular, only the theoretical fission cross-section evaluations at thermal point and fission resonance integral evaluations are known. It is not till now established experimentally, whether there is a threshold in fission reaction and, whether ^{233}Pa may be fissionable by neutrons. Under radiative capture of neutrons by ^{233}Pa two states of ^{234}Pa are to produce: g - the ground state with $T_{1/2}=6.70$ hours and m-metastable state with $T_{1/2}=1.17$ min. Production cross-sections of these states were obtained by integral methods only:

- in thermal range this is a cross-section averaged over Maxwell spectrum with the average energy 0.0253 eV (activation measurement on a beam removed from the reactor and passed through the thermal column);
- in resonance range this is a resonance integral obtained in activation measurements.

A summary capture cross-section at thermal point as a summary resonance integral agree with experimentally defined total capture cross-section in thermal point and total capture resonance integral. No differential measurements of capture cross-sections were carried out, but capture cross-sections, extracted from experimentally evaluated by beam transmission total cross-sections, were given only for the energies below 10 keV.

Among the modern evaluated neutron data libraries discussed by

us, only JENDL-3 and ENDF/B-6 contain the evaluations of characteristics of neutron and ^{233}Pa interaction.

T.Ohsawa, M.Inoue, T.Nakagawa (Kiuki Univ., Japan) are the authors of JENDL-3 evaluation (MAT 3913); 1987 is the year of evaluation completion. The range of resolved resonances is in the limits between 10^{-5} eV and 16.5 eV. Breit-Wigner formula for isolated levels is used in this region. There were used resonance parameters proposed in [6]. They were modified for reproduction of capture cross-section at thermal point and for capture resonance integral recommended at the same reference. The region of resolved resonances is between the limits 16.5 eV and 40 keV. Average resonance parameters proposed in [6] and also the values $S_0=0.75 \cdot 10^{-4}$, $S_1=1.5 \cdot 10^{-4}$, $\bar{D}=0.59$ eV, $\bar{\Gamma}_\gamma=0.047$ eV were used (S_1 value was fitted to reproduce the total cross-section and the capture cross-section about 20 keV from [Ta65]). Above 40 keV the capture cross-section was calculated by CASTHY code [Ig75]. In this case the values of $\bar{\Gamma}_\gamma=0.040$ eV and $\bar{D}=0.79$ eV from [Hu81] were used. Fission cross-section was calculated with the help of experimental data on fission probability from [Ga78]. The (n,2n) and (n,3n) reaction cross-sections was estimated by the model of [Se80].

ENDF/B-6 evaluation (MAT 9137) was carried out by Mathews, R.Kinsey, P.Joung (GA, BNL, LANL, USA) and completed in 1978. Data transferred from ENDF/B-5 were converted to ENDF/B-6 format. The region of resolved resonances was between the limits $1.0 \cdot 10^{-3}$ to 38.5 eV. Resonance parameters were taken from [Si67] for 28 resonances up to 17.5 eV. For 12 resonances from the range 17.5 eV - 37.5 eV data were derived from [Ha69]. The region of unresolved resonances is between the limits 38.5 eV to 10 keV. The resulting Γ_{n0} values for 40

resolved resonances and average $\bar{\Gamma}_{n0}$ values for the region of unresolved resonances were adjusted to yield of capture resonance integral from 0.5 eV to 10 MeV, which is equal to 859.96 barn. Above 10 keV capture cross-sections was defined by the following way: in the interval from 10 keV to 80 keV the cross-section was determined so as to transfer continuously to a cross-section calculated by the average resonance parameters; above 80 keV the cross-section of neutron capture by ^{233}Pa nucleus was equal to double ^{238}U capture cross-section from [St65]; normalization factor (=2) was chosen so that ^{233}Pa capture cross-section should be equal to ^{236}U capture cross-section at 0.9 MeV [St65]. Evaluation of the (n,2n) and (n,3n) reaction cross-sections was carried out according to the results from [Dr67]. Energy dependence of fission cross-sections was described as follows: at 0.48 MeV - 1 MeV ^{233}Pa fission cross-section was equal to ^{238}U fission cross-sections from [Da68]; at 1 MeV - 1.5 MeV natural logarithm of fission cross-section depended linearly on natural logarithm of energy; at 1.5 MeV - 5 MeV, $\sigma_f(^{233}\text{Pa}) = (C_1/C_2) \sigma_f(^{238}\text{U})$, where $\sigma_f(^{238}\text{U})$ was taken from [Da68]: $C_1 = 0.832$ was calculated by $\sigma_f(^{233}\text{Pa})$ on plateau from [La62]: $C_2 = 0.511$ was determined by average, for 2.6 MeV - 5.6 MeV interval, $\sigma_f(^{238}\text{U})$ value from [Da68]; at 5 MeV - 9 MeV energy dependence of ^{233}Pa fission cross-section was close to the same dependencies for $^{234,236}\text{U}$ and ^{237}U from the same reference; at 9 MeV - 12.5 MeV $\sigma_f(^{233}\text{Pa}) = (C_1/C_2) \sigma_f(^{238}\text{U})$, where $\sigma_f(^{238}\text{U})$ was taken from [Ha69], $C_1 = 1.56$ corresponded to the $\sigma_f(^{233}\text{Pa})$ value on the second plateau near 9 MeV, $C_2 = 1.02$ - corresponded to $\sigma_f(^{238}\text{U})$ value on the second plateau from [Ma69]; at 12.5 MeV - 15 MeV ^{233}Pa fission cross-section had the same energy variation as that for $^{236}\text{U}(n,f)$ one from [Ma69].

In Tables 3.10 - 3.14 the comparison of some characteristics of interaction between neutrons and ^{233}Pa nucleus calculated by JENDL-3 and ENDF/B-6 with recommended data and among themselves are given.

Table 3.10

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{233}Pa , in barn.

Value	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
σ_{f0}	0	0	< 0.1 < 0.1	1973, [5] 1984, [6]
$\sigma_{\gamma0}$	40.03	41.46	41^{+3}_{-3} $39.5^{+1.2}_{-1.2}$ $39^{+3.5}_{-3.5}$	1973, [5] 1984, [6] 1993, [3]

Table 3.11

The values of I_f , I_γ for ^{233}Pa , in barn.

Value	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
I_f	1.342	1.859	3.0	1984, [6]
I_γ (m+g)	864.3	859.6	895^{+30}_{-30} 860^{+35}_{-35} 897^{+23}_{-23}	1973, [5] 1984, [6] 1993, [3]

Table 3.12

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{233}Pa , in barn.

Value	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
JENDL-3	0.332	0.278	0.017	0.958	1.451	0.451
ENDF/B-6	0.481	0.179	0.007	1.589	0.520	0.210

Table 2.13

The values of ΔI_f , ΔI_γ for ^{233}Pa , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ΔI_f		ΔI_γ	
		JENDL-3	ENDF/B-6	JENDL-3	ENDF/B-6
0.5	10	0	0	687.1	685.4
10	10^2	0	0	120.0	116.3
10^2	10^3	0	0	39.49	40.30
10^3	10^4	0	0	11.61	12.42
10^4	10^5	0	0	3.790	3.620
10^5	10^6	0.086	0.003	2.080	0.380
10^6	10^7	1.256	1.856	0.210	0.210

Table 3.14

The values of $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{233}Pa , in barn.

$E_{\min}$ (eV)	$E_{\max}$	$\langle \sigma_f \rangle$		$\langle \sigma_\gamma \rangle$	
		JENDL-3	ENDF/B-6	JENDL-3	ENDF/B-6
0.5	10	0	0	189.5	188.5
10	10^2	0	0	44.36	42.72
10^2	10^3	0	0	12.31	13.95
10^3	10^4	0	0	4.061	4.406
10^4	10^5	0	0	1.560	0.991
10^5	10^6	0.078	0.003	0.630	0.309
10^6	10^7	0.692	1.024	0.040	0.045

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 ^{231}Pa

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4. Isotopes of $^{232-239}\text{U}$.

4.1 ^{232}U .

Even-even ^{232}U nucleus is a relatively long-lived one ($T_{1/2} = 72$ years) transforming into ^{228}Th due to α -decay. β - decay of ^{232}Pa ($T_{1/2} = 1.31$ days), α - decay of ^{236}Pu ($T_{1/2} = 2.85$ days) and the $^{233}\text{U}(n,2n)$ reaction are the channels of ^{232}U production. Main transmutation channels in neutron flux are mentioned below:

- fission and capture reactions in thermal and resonance neutron region;
- fission reactions in fission neutron region;
- fission reactions in thermonuclear neutron region, although the $(n,2n)$ reaction can be a competitive channel by some evaluations. The reaction of neutron capture by ^{232}U is an additional ^{233}U generation channel relative to the main $^{232}\text{Th}(n,\gamma)$ $^{233}\text{Th}(\beta^-)$ $^{233}\text{Pa}(\beta^-)$ ^{233}U one.

Among the evaluated neutron data versions compared by us the evaluations of characteristics of interaction between neutrons and ^{232}U are presented in JENDL-3 and ENDF/B-6.

T. Ohsawa (Kinki Univ., Japan) is the author of JENDL-3 (MAT 3921) evaluation. 1987 is a year of evaluation completion. The region of resolved resonances is between the limits 10^{-5}eV and 200eV . Recommended data from [6] were used to evaluate the resonance parameters. Reich-Moore representation from this work was changed to multilevel formula of Breit-Wigner type. Background cross-section was given to reproduce the results of fission cross-section measurements represented in [Au68] and [Fa70] at valleys of the levels. Above 200eV capture cross-section was calculated by CASTHY code [Ig75]. The $(n,2n)$ and $(n,3n)$ reaction cross-sections were calculated by

Segev-Fahima model [Se81]. Fission cross-section was calculated by using fission probabilities data of [Ga76] and compound nucleus production cross-section calculated by optical model. Below 1 keV cross-section was obtained using data from [Fa70].

ENDF/B-6 (MAT 9219) evaluation differs from ENDF/B-5 one by format of data representation and delayed neutron data. F. Mann is the author of ENDF/B-5 evaluation. 1977 is the year of evaluation completion. Recommended data from [Mu73] were used in the region of resolved resonances. Unlike Reich-Moore formula from mentioned work single-level Breit-Wigner formula for fission cross-section was used. Moreover, compared to the same work capture cross-section was slightly reduced at thermal point and fission one was increased but within the limits of error mentioned at the same work. Above resonance range fission cross-section was calculated by statistical model in a good agreement with data from [Au68] and [Fa70]]. Capture cross-section was calculated by the same model. Nothing was said about the evaluation method of the (n,2n) reaction cross-section in the head-line of MAT 9219 file.

Because of ^{232}U inaccessibility only a I_{γ} measurement result is known [Ha66], on which the recommended value from [5], preserved in more late recommendation from [6], is based. The only $I_f = 348 \pm 16$ barn [Gr76] measurement result overlapped by recommended values from [5,6] is also known.

In Tables 4.1-4.5 the comparison of characteristics of interaction between neutrons and ^{232}U among themselves and with recommended data is given.

Table 4.1

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{232}U , in barn.

Value	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
σ_{f0}	76.66	77.45	$75.2^{+4.7}_{-4.7}$	1973, [5]
			74^{+8}_{-8}	1978, [2]
			$76.8^{+4.8}_{-4.8}$	1984, [6]
$\sigma_{\gamma0}$	74.88	72.93	$73.1^{+1.5}_{-1.5}$	1973, [5]
			$74.9^{+1.6}_{-1.6}$	1984, [6]

Table 4.2

The values of I_f , I_γ for ^{232}U , in barn.

Value	JENDL-3	ENDF/B-6	Recomend. value	Year of public.
I_f	362.8	427.2	320^{+40}_{-40}	1973, [5]
	(364) ^{*)}		350^{+30}_{-30}	1984, [6]
I_γ	173.0	317.4	280^{+15}_{-15}	1973, [5]
	(173.0) ^{*)}		280^{+15}_{-15}	1984, [6]

*) - $E_{\text{max}} = 20 \text{ MeV}$

Table 4.3

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{232}U , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
JENDL-3	2.018	0.028	0.0008	2.370	0.150	0.005
ENDF/B-6	2.475	0.099	0.012	1.238	1.712	0.177

Table 4.4

The values of ΔI_f , ΔI_γ for ^{232}U , in barn.

$E_{\min}$ $E_{\max}$ (eV)		ΔI_f		ΔI_γ	
		JENDL-3	ENDF/B-6	JENDL-3	ENDF/
0.5	10	77.32	112.1	110.8	239.7
10	10^2	237.7	240.6	44.23	53.09
10^2	10^3	29.31	49.01	12.85	16.02
10^3	10^4	4.640	12.18	4.530	5.660
10^4	10^5	4.060	3.150	0.570	2.570
10^5	10^6	4.970	4.930	0.240	0.300
10^6	10^7	4.860	5.290	0.040	0.120

Table 4.5

The values of $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{232}U , in barn.&

$E_{\min}$ $E_{\max}$ (eV)		$\langle \sigma_f \rangle$		$\langle \sigma_\gamma \rangle$	
		JENDL-3	ENDF/B-6	JENDL-3	ENDF/B-6
0.5	10	43.58	66.23	66.59	147.2
10	10^2	62.76	66.68	10.64	17.49
10^2	10^3	6.919	17.06	5.262	5.433
10^3	10^4	1.710	3.713	1.481	2.203
10^4	10^5	1.975	1.340	0.205	0.793
10^5	10^6	2.000	2.415	0.034	0.196
10^6	10^7	2.222	1.995	0.008	0.041

4.2 ^{233}U .

Even-odd ^{233}U nucleus is a long-lived one ($T_{1/2}=1.592\cdot 10^5$ years, α -decay). It may be used as a fuel of power nuclear reactors. The $^{232}\text{Th}(n,\gamma)^{233}\text{Th}(\beta^-)^{233}\text{Pa}(\beta^-)^{233}\text{U}$ reaction chain is a main generation channel in breeder reactors, and the $^{232}\text{Th}(n,2n)^{231}\text{Th}(\beta^-)^{231}\text{Pa}(n,\gamma)^{232}\text{Pa}(\beta^-)^{232}\text{U}(n,\gamma)^{233}\text{U}$ reaction chain is a side channel. ^{233}U generation is possible both in thermal and fast neutron reactors, that distinguishes favorably Th-U cycle from U-Pu cycle, where only fast neutron reactor can be a breeder. Another expected advantage of Th-U cycle under ^{238}U absence in appreciable amounts in fuel content is a lesser contamination of waste fuel by actinides, as ^{232}Th is arranged on the bottom of actinide "sea". Fission reaction is a main ^{233}U transmutation channel in neutron flux of all energies. Radiation capture in resonance neutron spectrum and the $(n,2n)$ reaction in spectrum of thermonuclear ($E_n = 14$ MeV) neutrons are the additional transmutation channels. Characteristics of interaction of neutrons with ^{233}U nucleus have been studied intensively both experimentally and theoretically and their evaluations are given in all modern versions of evaluated neutron data compared by us.

R. Howerton (INLL, USA) is the author of ENDL-82 (MAT 7866); 1983 is a year of evaluation completion.

H. Matsunobu, Y. Kikuchi, T. Nakagawa (SAET, Japan) are the authors of JENDL-3 (MAT 3922) evaluation; 1987 is the a year of evaluation completion. The region of resolved resonances is between the limits

1 eV to 100 eV. Single-level Breit-Wigner formula is used. Evaluations of isolated level parameters are based on the results from [Ni74]. The region of resolved resonances is limited by the energy 30 keV from above. Average resonance parameters were deduced with the ASREP code [5] so that the cross-sections known for this energy region were reproduced. At the energy range below 1 eV capture cross-section evaluation was obtained according to the data from [We70], and fission cross-section evaluation of [Ca78], [De74], [Ps78]. At the energies above 30 keV fission cross-section evaluation is based on experimental data from [Gw76], [Ca78], [Ma87], [Ka85], [Iw87], [Me74], [Me87], [Po78] and on ^{235}U fission cross-section simultaneous evaluation [Ka86]. From 30 keV to 1 MeV capture cross-section was determined with the help of measurement results of α value from [Ho62] and JENDL-3 fission cross-section evaluation. Above 1 MeV capture cross-section was calculated by CASTHY (statistical model, [Ig75]) and ECIS (method of coupled channels, J.Raynal) codes with normalization by 0.0578 barn value at 1 MeV. The (n,2n) and (n,3n) reaction cross-sections were calculated by the method described in [Pe65], thereat the (n,2n) reaction cross-section was normalized to fission spectrum averaged value of 0.00408 barn [Ko73].

ENDF/B-6 (MAT 9222) cross-section evaluation differs from ENDF/B-5 one by format only. L. Stewart and et al. (LANL, USA) and L. Weston (ORNL, USA) are the authors of ENDF/B-5 evaluation; 1978 is a year of evaluation completion. The evaluation of N. Steen (BAPL, USA) is used in thermal range for fission and capture cross-sections. The region of resolved resonances extends to 60 eV. Multilevel formalism of Adler-Adler is used in contrast to ENDF/B-4 evaluation, where

single-level Breit-Wigner formula was used, due to which background fission cross-section exhibited strong fluctuations. The region of unresolved resonances extends to 10 keV. ENDF/B-4 version was represented in this range point by point. Re-evaluation of these data was started by F.Mann (MEDL, USA) using as initial results data of L.Weston (ORNL, USA), after which new evaluations of fission and capture cross-sections were obtained. Re-evaluation of fission cross-section was based on the results from [Ca78], [We68], [We70], [Ca70], [Ni74]. At re-evaluation of capture cross-sections data from [We70] were used. Much of data above 50 keV was obtained by model calculation since experimental information were insufficient or completely missing. Detail description of the used methods one can discover in [Ma78]. The (n,2n) and (n,3n) reaction cross-sections were calculated by statistical Hauser-Feshbach model [Ma78]; above 10 keV experimental data on capture cross-sections are missing, except measurement results of α value from [Ga76]. At extrapolation to 20 MeV the rise of capture cross-section due to direct process was accounted. Fission cross-section experimental data from 10 keV to 100 keV are sparse and reasonably discrepant. Above 100 keV there were used measurement results of $\sigma_f(^{233}\text{U})/\sigma_f(^{235}\text{U})$ ratio from [Ca78], [Me74] normalized by ENDF/B-5 evaluation of ^{235}U fission cross-section. The absolute data of [Po78] were also used, although they agreed unsatisfactorily with those mentioned immediately above. The evaluated curve was drawn as smooth as possible due to magnitude of the discrepancy among the sets.

A.B. Klepatskii, E.S.Sukhovitskii, V.M.Maslov, Yu.V.Porodzinskii, J.B. Morogovskii, L.A. Bakhanovich, A.M.Kolesov (IJE, Belorussia) are

the authors of BROND-2 (MAT 9211) evaluation; 1990 is a year of evaluation completion. The region of resolved resonances is limited by the energy 100 eV from above. Formalism of Adler-Adler is used. Resonance parameters were evaluated with the help of following experimental data: total cross-section from [Ko70]; fission cross-section from [De74], [Bl73]; capture cross-section from [We68]. The region of resolved resonances is limited by the energy 40 keV from above. Energy dependent resonance widths and level spacing (at $E_n=0.1$ keV, $\bar{\Gamma}_\gamma=32$ meV, $\bar{D}=0.42$ eV, both values for s-resonances) and also the average values of $S_0=0.99\cdot10^{-4}$, $S_1=1.54\cdot10^{-4}$, $R_0=9.6$ Fermi were used. At $E_n > 40$ keV capture cross-section evaluation was numerically carried out by statistical model from [Ig90]. The (n,2n) and (n,3n) reaction cross-sections and fission cross-sections were calculated by the same model. In last case data from [Po81], [Po83], [Gw76], [Fu78], [Ca78], [Du83] were used.

In Tables 4.6-4.12 the comparison of some characteristics of interaction between neutrons and ^{233}U nucleus obtained by ENDL-82, JENDL-3, ENDF/B-6 and BROND-2 evaluations among themselves and with recommend data are given.

Table 4.6

The values of σ_{f0} , $\sigma_{\gamma 0}$ for ^{233}U , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Yield of public.
σ_{f0}	525.1	529.9	528.5	529.6	$531.1^{+1.3}_{-1.3}$	1973, [5]
					511^{+17}_{-17}	1978, [2]
					$529.1^{+1.2}_{-1.2}$	1982, [7]
$\sigma_{\gamma 0}$	45.90	45.30	45.76	45.40	$47.7^{+2.0}_{-2.0}$	1973, [5]
					$45.5^{+0.7}_{-0.7}$	1982, [7]

$E_{\min}$ strong dependence of resonance integrals takes place there. Thus at $E_{\min} = 3$ eV experimental values of resonances integrals, presented in [8], are equal to the following values: $I_f = 405 \pm 24$ barn and $I_f = 60 \pm 3$ barn, and for $E_{\min} = 0.55$ eV from [2] the value $I_f = 812 \pm 90$ barn measured in [Gr76] is given.

Table 4.7

The values of I_f , I_γ for ^{233}U , in barn.

Value	ENDL-82	JENDL-3	ENDF/B -6	BROND-2	Recommend. value	Year of & public.
I_f	753.1	770.6 (772)	755.8	792.5 (779.9) **)	764^{+13}_{-13}	1973, [5]
					750^{+20}_{-20}	1973, [2]
					760^{+17}_{-17}	1984, [6]
I_γ	133.8	138.6 (139)	136.8	142.0 (139.5) **)	140^{+6}_{-6}	1973, [5]
					140^{+10}_{-10}	1973, [2]
					137^{+6}_{-6}	1984, [6]

*) - $E_{\max} = 20$ MeV

**) - $E_{\max} = 100$ keV

Table 4.8

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{233}U , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.896	0.060	0.008	2.250	0.470	0.050
JENDL-3	1.942	0.064	0.004	2.348	0.200	0.018
ENDF/B-6	1.905	0.054	0.008	2.278	0.511	0.003
BROND-2	1.867	0.059	0.005	2.265	0.411	0.005
Measured value	$1.852^{+0.036}_{-0.036}$, [9] $1.893^{+0.048}_{-0.048}$, [10,11]					

Table 4.9

The value of ΔI_f for ^{233}U , in barn.

$E_{\min}$ $E_{\max}$ (eV)		ΔI_f			
		ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	515.0	528.8	519.2	539.4
10	10^2	162.2	168.0	163.0	173.2
10^2	10^3	43.12	42.85	41.50	48.62
10^3	10^4	16.44	15.23	16.19	15.91
10^4	10^5	7.190	6.140	7.060	6.930
10^5	10^6	4.840	5.110	4.430	4.270
10^6	10^7	4.280	4.440	4.320	4.250

Table 4.10

The value of ΔI_{γ} for ^{233}U , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	101.5	106.2	103.5	105.8
10	10^2	23.23	24.61	24.12	26.34
10^2	10^3	6.050	5.680	5.860	6.770
10^3	10^4	1.770	1.530	2.150	1.940
10^4	10^5	0.480	0.510	0.360	0.390
10^5	10^6	0.700	0.570	0.760	0.710
10^6	10^7	0.050	0.060	0.030	0.050

Table 4.11

The value of $\langle \sigma_f \rangle$ for ^{233}U , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	131.3	135.8	132.6	139.0
10	10^2	50.63	52.95	51.13	54.90
10^2	10^3	15.46	15.49	15.02	17.17
10^3	10^4	6.003	5.511	5.896	5.643
10^4	10^5	2.770	2.571	2.606	2.550
10^5	10^6	2.027	2.005	2.014	1.970
10^6	10^7	1.898	1.959	1.915	1.893

Table 4.12

The value of $\langle \sigma_{\gamma} \rangle$ for ^{233}U , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	27.67	29.48	28.65	29.20
10	10^2	7.077	7.002	6.962	8.869
10^2	10^3	1.975	1.966	2.107	2.266
10^3	10^4	0.672	0.511	0.779	0.714
10^4	10^5	0.292	0.280	0.283	0.304
10^5	10^6	0.142	0.133	0.138	0.134
10^6	10^7	0.012	0.012	0.006	0.012

4.3 ^{234}U .

Even-even ^{234}U nucleus is a long-lived one ($T_{1/2} = 2.45 \cdot 10^5$ years, α -decay). The ^{233}U (n, γ) and ^{235}U ($n, 2n$) reactions are the main channels of ^{234}U generation in nuclear reactors. ^{238}Pu (α) ($T_{1/2} = 87.74$ years), ^{234}Pa (β^-) ($T_{1/2} = 6.75$ hours), $^{234\text{m}}\text{Pa}$ (β^-) ($T_{1/2} = 1.17$ min) are the additional channels. Small efficiency of two last channels is due to the fact, that ^{233}Pa parent nucleus, during radiative capture of which ^{234}Pa ground and isomeric states are produced, has a small period of β^- decay ($T_{1/2} = 27$ days). Capture reactions in thermal and resonance neutron regions and fission reactions in fission and thermonuclear ones are the main transmutation process of ^{234}U nucleus in neutron flux:

Because of ^{234}U inaccessibility the results of resonance integral measurement are missing, although the evaluations of characteristics

of interaction of neutrons with this nucleus are presented in most versions of evaluated neutron data compared by us.

R.Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7867) evaluation. 1987 is a year of evaluation completion.

T.Watanabe (Kawasaki, Japan) is the author of JENDL-3 (MAT 3922) evaluation. 1987 is a year of evaluation completion. The region of resolved resonances is between the limits 10^{-5} eV and 1.5 keV. Multilevel Breit-Wigner formula is used. Resonance parameters from [Ja77] after modification of an average radiative width to 0.026 eV [6] are used. For reproduction of recommended in [6] cross-sections at thermal point the resonance level at negative energy of 2.06 eV was added. The region of unresolved resonances is limited by the energy 50 keV from above. The average radiative width and level spacing were taken from [6] and were equal to the following values $\bar{\Gamma}_{\gamma} = 0.026$ eV, $\bar{D} = 10.6$ eV. Average fission width is zero but non-resonance fission cross-section is non-zero. In addition to the following average parameters were used: $S_0 = 0.964 \cdot 10^{-4}$ (calculated by the method of coupled channels according to ECIS code [Ra70]): $S_1 = 1.197 \cdot 10^{-4}$ (fitted to results of total cross-section calculations by ECIS code); $R = 9.70$ Fermi (fitted to total cross-section at 50 keV). At $E > 50$ keV capture cross-sections were calculated by ECIS and CASTHY codes (optical statistical model with spherical optical potential [Ig73]). Parameters of deformed optical potential for ECIS code were taken from [La82]. Parameters of spherical optical potential for CASTHY code were fitted so as to reproduce the total cross-section calculated by ECIS code. The (n,2n) and (n,3n) reaction cross-section evaluations from JENDL-2 were renormalized so as to agree with

results of calculations of fission cross-section and production of compound nucleus according to the ECIS and CASTHY codes. In evaluating fission cross-section the results on measurement of ratio between ^{234}U fission cross-section and ^{235}U fission cross-section from [Be77], [Me78] and [Ka86] were used. The evaluation of ^{235}U fission cross-section from [Ja77] was also used.

ENDF/B-6 evaluation differs from ENDF/B-5 one by format only. M. Divadeenam, F. Mann, M. Drake, et al. (BNL, MEDL, GGA, USA) are the authors of ENDF/B-5 evaluation. 1978 is a year of evaluation completion. The region of resolved resonances is limited by the energy 1.5 keV from above. Resonance parameters were taken from [Ja77]. Bound level parameters were modified for agreement with thermal cross-sections and resonance integrals recommended in [Mu73]. The region of unresolved resonances is limited by the energy 100 keV above. Average resonance parameters were obtained by averaging over fission cross-section from [Ja77] and ENDF/B-4. The capture cross-sections from 1.5 keV to 100 keV were calculated by code UR (U. Pennington, private communication). Above 100 keV capture cross-section evaluation from [Dr67] was used. The (n,2n) and (n,3n) reaction cross-sections were evaluated according to results of [Pa64]. In evaluating fission cross-section the measurement results of $\sigma_f(^{234}\text{U})/\sigma_f(^{235}\text{U})$ ratio from [Be 75] and ENDF/B-5 evaluation of ^{235}U fission cross-section were used.

In Tables 4.13-4.17 characteristics of interaction between neutrons and ^{234}U nucleus corresponded to ENDL-82, JENDL-3 and ENDF/B-6 evaluations are compared among themselves and with recommended values.

Table 4.13

The value of σ_{f0} , $\sigma_{\gamma0}$ for ^{234}U , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
σ_{f0}	0	0.006	0.464	<0.65	1973, [5]
				<0.65	1984, [6]
$\sigma_{\gamma0}$	99.63	99.75	103.1	$100.2^{+1.5}_{-1.5}$	1973, [5]
				$99.8^{+1.3}_{-1.3}$	1984, [6]

Table 4.14

The values of I_f , I_γ for ^{234}U , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
I_f	4.507	5.264	5.113	6.5	1984, [6]
		(6.720) ^{*)}			
I_γ	94.16	631.7	660.8	630^{+70}_{-70}	1973, [5]
		(632.0) ^{*)}		660^{+70}_{-70}	1984, [6]

*) - $E_{\text{max}} = 20 \text{ MeV}$

Table 4.15

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{234}U , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.231	0.136	0.009	2.121	0.250	0.300
JENDL-3	1.226	0.213	0.007	2.067	0.473	0.127
ENDF/B-6	1.233	0.170	0.002	2.059	0.382	0.150
Measured value	$1.204^{+0.014}_{-0.014}$					

Table 4.16

The values of ΔI_f , ΔI_γ for ^{234}U , in barn.

$E_{\min}$ (eV)	$E_{\max}$		ΔI_f			ΔI_γ		
			ENDL-82	JENDL-3	ENDF/B-6	ENDL-82	JENDL-3	ENDF/B-6
0.5	10	0	0.407	0.311	59.20	538.11	568.7	
10	10^2	0	0.025	0.016	17.56	60.76	56.14	
10^2	10^3	0	0.318	0.227	10.02	26.04	28.13	
10^3	10^4	0	0.047	0.024	4.864	4.600	5.320	
10^4	10^5	0	0.045	0.039	1.758	1.550	1.690	
10^5	10^6	0.916	0.873	0.945	0.606	0.320	0.570	
10^6	10^7	3.591	3.549	3.549	0.156	0.340	0.210	

Table 4.17

The values of $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{234}U , in barn.

$E_{\min}$ (eV)	$E_{\max}$		$\langle \sigma_f \rangle$			$\langle \sigma_\gamma \rangle$		
			ENDL-82	JENDL-3	ENDF/B-6	ENDL-82	JENDL-3	ENDF/B-6
0.5	10	0	0.220	0.144	15.41	285.8	301.6	
10	10^2	0	0.017	0.012	6.605	32.22	31.36	
10^2	10^3	0	0.154	0.111	4.135	7.525	8.299	
10^3	10^4	0	0.021	0.013	1.706	1.528	1.788	
10^4	10^5	0	0.018	0.019	0.620	0.464	0.546	
10^5	10^6	0.661	0.647	0.686	0.247	0.246	0.301	
10^6	10^7	1.709	1.671	1.677	0.035	0.080	0.049	

4.4 ^{235}U .

Even-even ^{235}U nucleus is a long-lived one ($T_{1/2}=7.038\cdot 10^8$ years α - decay) of natural origin (0.72% in natural mixture of uranium nuclei). Nucleus of ^{235}U is a main energy release component of fuel of modern power reactors. In the absence of thorium ^{235}U mainly burns out in fission reactions, although the $^{235}\text{U} (n,2n) ^{234}\text{U} (n,\gamma) ^{235}\text{U}$ closed chain is possible in neutron spectrum. With thorium in fuel content weak accumulation of ^{235}U is possible due to the $^{232}\text{Th}(n,\gamma) ^{233}\text{Th}(\beta^-) ^{233}\text{Pa}(\beta^-) ^{233}\text{U}(n,\gamma) ^{234}\text{U}(n,\gamma) ^{235}\text{U}$ or : $^{233}\text{Pa}(n,\gamma) ^{234}\text{Pa}(\beta^-) ^{234}\text{U}(n,\gamma) ^{235}\text{U}$ reaction chains. ^{235}U is a fissionable by neutrons of all energies nucleus.

Characteristics of interaction between neutrons and ^{235}U nucleus have been studied in detail both theoretically and experimentally for about fifty of years. Their evaluations are represented in all versions of evaluated neutron data discussed by us.

R.Howerton is the author of ENDL-82 (MAT 7868) evaluation. 1983 is a year of evaluation completion.

L.Weston, P. Young, W. Poenits (ORNL, LANL) are the authors of ENDF/B-6 (MAT 9228) which was completed 1989. In the thermal neutron region most weight was given to measurement results from the following works: neutron transmission [Sp87]; the recent measurements of fission cross-sections [Gw84], [Sc83], [Wa88]. ENDF/B-5 evaluation of capture cross-section below 0.5 eV was renormalized by cross-section value at thermal point proposed by ENDF/B-6 Standards Committee. Range of resonance description extends to 2250 eV. Reich - Moore representation is used. Total number of resonances accounted is equal to 3343. The range is divided into ten regions, each being

described by its own set of resonance parameters. Resonance parameters were determined by SAMMY code [La85] according to best agreement with experimental data which were renormalized by the values in thermal point recommended by ENDF/B-6 Standards Committee.

In the range from 0 to 110 eV most significance was attached to experimental results from the following works: the neutron transmission - [Ha83]; the fission cross-section measurements - [Gw84], [Sc83], [Mo78] and [We84]. Capture and fission cross-section evaluations lead to good agreement with measurements; results of average, over some intervals, cross-sections from [Sa67], [Pe73] and [Sc83], [Gw84], [Wa85].

Above 110 eV the resonances could not be fully resolved. Evaluations of resonance parameters were carried out using experimental results on neutron transmission from [Ha88] and on fission cross-section measurements from [We84] and [We88]. Due to evaluation interval average fission cross-sections at 100-2000 eV are in good agreement with proposed ones by ENDF/B-6 Standards Committee. Nevertheless, the ratio between capture and fission resonance integrals equal to 0.477 is less than the value 0.513 ± 0.015 , measured directly in [Ha79] (in ENDF/B-5 evaluation this value was equal to 0.493).

The region of unresolved resonances is limited by the energy 25 keV from above. When obtaining of the average resonances parameters the recommendations of Standard Committee for the fission cross-sections and new capture cross-section evaluations based on newer α -measurements (Supplement ANL-83-4) were accounted. Above 100 keV the evaluations were based on a careful theoretical analysis

utilizing the available experimental data. Following codes were used under the calculations: ECIS [Ra70] for coupled channel optical model; GNASH [Ar88], [Yo77] and COMNUC [Du70] code systems for Hauser-Feshbach statistical model with preequilibrium emission and fission. According to GNASH code the (n,2n) and (n,3n) reaction cross-sections were calculated. For fission cross-sections from 2.2 keV to 120 keV the average values from evaluation of W. Foenitz and ENDF/B-5 one, were used. From 100 keV to 20 MeV the Standards Committee recommendations were used. From 2.25 keV to 1 MeV in capture cross-section evaluation the results of newer α - measurements (Supplement ANL 83-4) and Standards Committee recommendations relative to ^{235}U fission cross-section were used. From 1 MeV to 20 MeV capture cross-section was calculated by COMNUC/GNASH codes with addition of contribution of semi-direct component above few MeV.

BROND-2 (MAT 9241) evaluation was carried out in 1985 year in IJE (Belorussia) by V.A. Kon'shin, G.V. Antsipov, E.S. Sukhovitskii, A.W.Klepatskii, V.M. Maslov and G.B. Morogovskii. The region of resolved resonances is between the limits 0.29 and 99.5 eV and contains 205 s-resonances. For calculation of energy dependence of cross-sections multilevel Breit-Wigner formula is recommended. Formalism of Adler-Adler don't lead to result proving in comparison with formula mentioned above. Resonance parameters were determined using following experimental data: for total cross-section from [Sh58], [Mi65]; for fission cross-section from [Mo78], [De71], [Ca75], [B171]; for capture cross-section from [De66], [Pe73]. Spin identification of levels was performed from the data of [Mo78]. A background cross-section ("smooth file") should be added to the

cross-sections calculated from the resonance parameters. The cross-section calculation from resonance parameters was carried out in the 5-100 eV region. From 10^{-5} eV to 5 eV the cross-sections were given point by point. In the $0.1 \cdot 10^{-5}$ -1 eV region there were used following recommended data: for total cross-section from [Sa59], [Bl60], [Si60], [Le55], [Sh58]; for fission cross-section from [De71], [Sa55], [Bo59], [Sh58], [De66]; for capture cross-section from [Gw76]; for elastic scattering cross-section calculated from the resonance parameters.

At the energy range of unresolved resonances (0.1-100 keV) evaluated data were obtained on the base of following experimental results: for total cross-section from [Mi65], [Ut67], [Po79]; for fission cross-section from [Sa66], [Gz76], [Wa76], [Gw76], [Bl73], [Pe73], [Mi65], [Ga72], [Sz71], [Po74]; capture for cross-sections from [Mu80], [Po76], [Pe73], [Gz70]. Following average resonance parameters providing for fluctuations of cross-sections in the range up to 25 keV were obtained according to evaluated cross-section data:

$$\bar{\Gamma}_{\gamma} = 31 \text{ meV (the same for s and p waves),}$$

$$S_0 = 0.98 \cdot 10^{-4} \text{ (over the whole unresolved resonance region),}$$

$$S_1 = 1.6 \cdot 10^{-4} \text{ (over the whole unresolved resonance region).}$$

The contribution of the (n, γ f) process to the fission cross-section is 2.4% in the range of 0.1 - 100 keV, and corresponding cross-section was added to the total fission cross-section. The difference between the results of calculations from average parameters and the evaluated cross-sections are given in the form of smooth backgrounds to cross-sections and accounted for, in absolute value: in the range 0.1-30 keV, 0.5-1.5% -for fission and capture cross-sections, 3-5% for

total cross-section; in the range 30-100 keV 5.4% for fission cross-section, 4% for capture cross-section, 1.7% for total cross-section. At the range 0.1-20 MeV total cross-section evaluation is based both on experimental data and on calculations by coupled channel method. The (n,2n) and (n,3n) reaction cross-sections were calculated by the multiscascade statistical model. Fission barriers and level density parameters in terms of collective effects were obtained from analysis of cross-sections of uranium isotope fission in the region of the "first plateau". Fission cross-section evaluation was carried out by the method of correlation account between the errors from various experiments on the base of analysis of available experimental data. Evaluated data agree with the ENDF/B-5 evaluation to within 1-2% (in the range 0.4-5 MeV they, on the average, are 1% lower than the data from ENDF/B-5). The errors of the evaluated fission cross-sections were 3% on an average, except for the narrow region around 14 MeV, where ENDF/B-5 evaluation is adopted. Capture cross-section evaluation was carried out with the help of correlation account of various data with α -measurement results. Experimental data from [Mu80] and [Po76] are considered to be determining ones. Data from another references were accounted with a lower "weight". An error in evaluated values accounted for 4% at $E_n < 30$ keV and 8-10% at $0.03 < E_n < 1$ MeV.

H. Matsunobu, K.Hida, T.Nakagawa and Y.Nakajima (JAERI, SAEI, NAIG, JAPAN) are the author of JENDL-3 evaluation. 1987 is a year of evaluation completion. The region of resolved resonances is between the limits 1 and 100eV. Double products of total width into neutron one were evaluated by simple averaging of experimental data. Radiation

widths were evaluated by averaging of experimental data with weights. Fission widths were calculated by the area under fission resonances. The details one can find in [Na81]. Identification of level spins was carried out according to the data from [Mo78]. The region of unresolved resonances is between the limits 100 eV and 30 keV. The evaluation of total, capture and fission cross-sections was carried out by fitting with varying of S_0 and S_1 values, average fission and radiation widths. The fission cross-section was based on measurements from [We84]. The capture cross-section was determined as a product of fission cross-section into JENDL-2 evaluation of α value. Below 1 eV evaluations of cross-sections, given point by point, are based on experimental data. At $E_n > 30$ keV capture cross-section evaluation up to 1 MeV was carried out by multiplying of evaluated fission cross-section into α -measurement results from [Ko75], [Dw76], [Gw76], [B176], [Ho62], [Be79], [Go82]. At $E_n > 1$ MeV capture cross-section was calculated by CASTHY code [Ig75]. The (n,2n) and (n,3n) reaction cross-sections were calculated by estimations on evaporation model using experimental data: from [Fr80] for (n,2n), from [Ve78] for (n,3n) and (n,4n). Fission cross-section evaluation was carried out in [Ka85] and [Ka86] on the basis of the capture cross-sections of ^{197}Au and ^{238}U , the fission cross-sections of $^{235,238}\text{U}$ and $^{239-241}\text{Pu}$ in the energy range from 50 keV to 20 MeV. There were used experimental data from following references: [We84], [Pe74], [Po74], [Po77], [Gz75], [Sz76], [Ba76], [Ca78b], [Ca81], [Ca78], [Ka78], [Ad79], [Ar79], [Ar83], [Wa82], [Li82], [Ma82], [Ca82], [Co82], [Du84].

In Tables 4.18-4.24 the comparison of characteristics of

interaction between neutrons and ^{235}U , calculated by ENDL-82, JENDL-3, ENDF/B-6 and BROND-2 evaluations, among themselves and with recommended values are given.

Table 4.18

The values of σ_{f0} , σ_{f1} for ^{235}U , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend& value	Year of public.
σ_{f0}	602.0	584.0	584.6	582.6	$582.2^{+1.3}_{-1.3}$	1973, [5]
					$582.6^{+1.1}_{-1.1}$	1982, [7]
$\sigma_{\gamma 0}$	100.9	96.0	98.8	98.3	$98.6^{+1.5}_{-1.5}$	1973, [5]
					$98.3^{+0.8}_{-0.8}$	1982, [7]

Table 4.19

The values of I_f , I_g for ^{235}U , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	282.6	273.5	280.7	281.2	275^{+5}_{-5}	1973, [5]
		(275.0) ^{*)}		(270.0) ^{**))}	275^{+5}_{-5}	1984, [6]
I_{γ}	139.8	152.0	133.2	145.6	144^{+6}_{-6}	1973, [5]
		(152.0) ^{*)}		(143.0) ^{**))}	144^{+6}_{-6}	1984, [6]

*) - $E_{\text{max}} = 20 \text{ MeV}$

**) - $E_{\text{min}} = 100 \text{ keV}$

Table 4.20

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{235}U , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.232	0.091	0.016	2.046	0.363	0.074
JENDL-3	1.235	0.087	0.016	2.088	0.543	0.042
ENDF/B-6	1.220	0.092	0.016	2.060	0.504	0.038
BROND-2	1.237	0.089	0.018	2.068	0.568	0.042
Measured value				$2.083^{+0.015}$		[13]

Table 4.21

The value of ΔI_f for ^{235}U , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ΔI_f			
		ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	123.6	119.5	107.8	121.2
10	10^2	98.78	94.29	102.2	92.96
10^2	10^3	37.26	37.45	39.05	43.05
10^3	10^4	12.05	11.65	18.50	12.66
10^4	10^5	5.360	5.140	7.400	5.090
10^5	10^6	2.630	2.550	2.780	3.220
10^6	10^7	3.010	2.990	2.940	2.990

Table 4.22

The value of ΔI_γ for ^{235}U , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ΔI_γ			
		ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	48.43	52.24	40.03	49.20
10	10^2	64.55	73.66	66.36	71.98
10^2	10^3	19.53	18.56	16.47	17.22
10^3	10^4	4.720	5.130	7.280	4.710
10^4	10^5	1.770	2.000	2.660	1.890
10^5	10^6	0.700	0.340	0.270	0.550
10^6	10^7	0.080	0.070	0.090	0.080

Table 4.23

The value of $\langle \sigma_f \rangle$ for ^{235}U , in barn.

$E_{\min}$ (eV)	$E_{\max}$	$\langle \sigma_f \rangle$			
		ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	42.95	41.57	33&24	43.41
10	10^2	37.89	37.45	39.68	37.21
10^2	10^3	13.26	13.38	14.15	15.13
10^3	10^4	4.394	4.142	7.575	4.627
10^4	10^5	1.941	1.877	2.313	1.944
10^5	10^6	1.200	1.200	1.196	1.220
10^6	10^7	1.399	1.385	1.369	1.383

Table 4.24

The values of $\langle \sigma_{\gamma} \rangle$ for ^{235}U , in barn.

$E_{\min}$	$E_{\max}$	$\langle \sigma_{\gamma} \rangle$			
(eV)		ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	24.70	28.03	18.64	25.09
10	10^2	20.69	24.82	21.97	23.43
10^2	10^3	7.551	6.240	5.667	5.867
10^3	10^4	1.554	1.785	2.812	1.652
10^4	10^5	0.708	0.639	0.734	0.668
10^5	10^6	0.192	0.193	0.181	0.189
10^6	10^7	0.020	0.012	0.018	0.015

^{235}U neutron fission cross-section is generally recognized standard during the pursuance of neutron research. Data on ^{235}U fission cross-section from ENDF/B-5 (MAT 1395) evaluation were included into International file of standard neutron cross-sections. In spite of great attention focusing on cross-section evaluation of this reaction, needed accuracy ($\pm 1\%$) over the whole energy range except the region near 14 MeV, has not been reached. Thus ^{235}U fission cross-section from ENDF/B-6 evaluation agrees with ENDF/B-5 evaluation only at $E_n > 4$ MeV, for the range $1 \text{ MeV} < E_n < 4 \text{ MeV}$ ENDF/B-6 evaluation is 1-2% below than ENDF/B-5 evaluation and at $0.1 \text{ MeV} - 1 \text{ MeV}$ it is 2% below. For completeness Table 4.25 is listed, where calculated ^{235}U fission cross-sections values averaged over neutron spectrum of ^{252}Cf spontaneous fission are compared with measurement results.

Table 4.25

The value of σ_f for ^{235}U , in barn.

ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Measurem. results	Year of public.
1.232	1.235	1.220	1.237	$1.204^{+0.029}_{-0.029}$	1975, [14]
				$1.205^{+0.027}_{-0.027}$	1976, [15]
				$1.215^{+0.022}_{-0.022}$	1978, [16]
				$1.204^{+0.024}_{-0.024}$ *)	1981, [17]
				$1.205^{+0.025}_{-0.025}$	1982, [18]
				$1.208^{+0.017}_{-0.017}$	1982, [19]
				$1.216^{+0.019}_{-0.019}$	1984, [11]
				$1.234^{+0.017}_{-0.017}$	1985, [20]

*) - recommended value

4.5 ^{236}U .

Even-even ^{236}U nucleus is a long-lived one $T_{1/2} = 2.3415 \cdot 10^7$ years, α -decay). The $^{235}\text{U}(n, \gamma)$ reaction, going more actively in thermal and resonance neutron flux, is the main generation channel in nuclear reactors. Capture reaction with production of ^{237}U nucleus ($T_{1/2} = 6.75$ days), which according to β^- - decay transforms into ^{237}Np , is the main transmutation channel in thermal and resonance neutron flux. Process of ^{236}U fission is more probable in fission and thermonuclear neutron flux.

Because of ^{236}U inaccessibility characteristics of interaction of neutrons with this nucleus have been studied noticeably worse than in case of ^{235}U . Nevertheless, the evaluations of these characteristics are presented in all versions of evaluated neutron data compared by us. This interest is first due to the fact, that ^{236}U nucleus is one of ^{235}U transmutation product on the channel of radiation neutron capture, and second, the knowledge of these characteristics is necessary for justified evaluation of ^{237}Np generation at ^{235}U depletion by the $^{235}\text{U}(n, \gamma) \rightarrow ^{236}\text{U}(n, \gamma) \rightarrow ^{237}\text{U}(\beta^-) \rightarrow ^{237}\text{Np}$ reaction chain.

R. Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7869) evaluation. 1983 is a year of evaluation completion.

F. Mann, R. Schenter (MEDL, USA) are the authors of ENDF/B-6 evaluation. 1987 is a year of evaluation completion. The region of resolved resonances is between the limits 10^{-5} eV and 1.5 keV. New data from [Ma89] were used above 20 eV. For resonances at - 9.7 eV and 5.45 eV data from [6] were used. Data from [Ma89] were combined with resonance parameters from [6]. The last were evaluated with

measurement results from [Ca76], [Me75], [Ca76]. Resonance energies and fission widths, which are quite small, were also taken from [6]. Capture and fission cross-sections were completely defined by resonance parameters.

The region of unresolved resonances is limited by the energy 100 keV from above and the cross-sections are described by average resonance parameters of s-waves obtained by evaluations at the region of resolved resonances and fitting (effective account of p and d - waves) to recent results of capture and absorption cross-section measurements from [Ka85], [Be82], [Da85], [Ca71], [Ba61]. FERRET code from [Sc79] was used in fitting procedure. Energy dependence of widths of channels competing with capture process was the same as for ^{238}U from ENDF/B-5 evaluation.

Above 100 keV fission cross-section evaluation was carried out according to the data from [Be75], [Ma76] normalized on ^{235}U fission cross-section from ENDF/B-6 evaluation. The (n,2n) and (n,3n) cross-sections were taken from [Pa64] and [Dr67]. Above 100 keV to 20 MeV multigroup capture cross-section values were obtained by subtracting of previous ENDF/B-5 fission cross-sections from total absorption cross-section of [Ma89]. A smooth cross-section curve was obtained by combination of data from [Ma89] with recent experimental data from [Ka85], [Be82], [Da85] and older data from [Ba61]. FERRET code was used in this case. The shape of the capture curve above 1 MeV was assumed to be nearly the same as for ^{238}U .

BROND-2 (MAT 9251) evaluation was completed in 1986 and was carried out by A.B. Klepatskii, V. A. Kon'shin, V. M. Maslov, Yu.V.Porodzinskii, E. Sh. Sukhovitskii (IJE, Belorussia). The region

of resolved resonances is between the limits 10^{-5} and 1 keV. Multilevel formalism of Breit-Wigner is used. Determination of resonance parameters of 72 levels is based on using of experimental data from [Ha58], [Ca76], [Me75], [Ca70], [Th72], [Ha69].

The region of unresolved resonances is between the limits 1 and 150 keV. Evaluated average resonance parameters were obtained by experimental data from [Be82], [Ka85], [Gr83], [Da85], [Ca70]. Average radiative and fission widths depending on energy and level spacing were determined. At $E_n > 150$ keV capture cross-sections was evaluated by experimental data from [Gr83], [Da85], [To85], [St61], [Ba61]. Preference is given to results from the first three works, and the data evaluated by us lie 40% below than the data from two last works. Fission cross-section evaluation at the same energy range was mainly based on experimental data from [Be77], [Me78], [Go84], [Fu85], [La55]. The $(n,2n)$ and $(n,3n)$ reaction cross-section were calculated by cascade statistical model. Fission barriers and parameters of level density for transition states were obtained from fission cross-section analysis of uranium isotopes in the range of "first plateau".

JENDL-3 (MAT 3925) evaluation was completed by T.Yoshida (NAIG, Japan) in 1988. T. Yoshida carried out the evaluation of JENDL-2 in 1979. This evaluation was critically revised by him in 1986 and in 1988 JENDL-3 evaluation was assembled on this base. There were accounted new results of capture cross-section measurements obtained in Russia during 1982-1986 years, as a result of which the decrease of capture cross-section (in comparison with JENDL-2 evaluation) by approximately 3% (above 1.5 keV) was performed. From 1.5 keV to 700 keV underthreshold fission cross-section was introduced. Unknown

fission width is adopted to be equal to 0.354 meV. The region of resolved resonances is between the limits 10^{-5} eV and 1.5 eV. Multilevel Breit-Wigner formula is used. Resonance energies and Γ_n (at $\Gamma_n > 0.1 \Gamma_\gamma$) corresponded to data from [Ca76]. At $\Gamma_n < 0.1 \Gamma_\gamma$ the values of Γ_n were evaluated according to data from [Me75]. Γ_γ values were evaluated on data from work of the same authors, in the absence of corresponding data Γ_γ value was assumed to be equal to Γ_f . Fission widths were determined according to the data from [Te72], average values of radiative and fission widths were equal to 23 meV and 0.354 meV, respectively. Resonance at negative energy was adopted for capture cross-section reproduction in thermal point recommended in [6]. The region of unresolved resonances is limited by the energy 40 keV from above. Average resonance parameters were defined so as to reproduce both total and capture cross-sections calculated by statistic model with the help of CASTHY code [Ig75] and evaluated fission cross-section. In the region of resolved resonances background cross-sections is zero for all processes. In the region of unresolved resonances only background cross-section for scattering is different from zero. Above 40 keV capture cross-section was calculated by statistical model with the help of CASTHY code. The (n,2n) and (n,3n) reaction cross-sections were calculated by PEGASUS code [Ii87]. Fission cross-section was evaluated on measurements results of ratio of ^{236}U fission cross-section to ^{235}U fission cross-section using JENDL-3 evaluation of ^{235}U fission cross-section.

In Tables 4.26-4.32 calculated values of characteristics of interaction of neutrons and ^{236}U nucleus on ENDL-82, JENDL-3, ENDF/B-6 and BROND-2 are compared among themselves and with recommended data.

Table 4.26

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{236}U , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
σ_{f0}	0.500	0.061	0.047	0.068	0	1970, [21]
					0.07	1984, [6]
$\sigma_{\gamma0}$	5.000	5.295	5.132	5.000	$5.2^{+0.3}_{-0.3}$	1973, [5]
					$5.11^{+0.21}_{-0.21}$	1984, [6]

Table 4.27

The values of I_f , I_γ for ^{236}U , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	4.661	6.619	6.681	6.674	0	1970, [21]
		(7.77) ^{*)}		(4.44) ^{**)}	$7.8^{+1.6}_{-1.6}$	1984, [6]
I_γ	831.3	345.9	342.1	332.2	365^{+20}_{-20}	1973, [5]
		(346.0) ^{*)}		(331.1) ^{**)}	360^{+15}_{-15}	1984, [6]

*) - $E_{\max} = 20 \text{ MeV}$

**) - $E_{\max} = 100 \text{ keV}$

Table 4.28

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{236}U , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	0.588	0.168	0.019	1.610	0.250	0.700
JENDL-3	0.602	0.105	0.011	1.659	0.487	0.523
ENDF/B-6	0.592	0.091	0.012	1.628	0.550	0.550
BROND-2	0.594	0.089	0.010	1.582	0.855	0.165
Measured value	$0.612^{+0.008}_{-0.008}$, [12]					

Table 4.29

The value of ΔI_f for ^{236}U , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	1.498	3.249	3.356	3.369
10	10^2	0.854	0.748	0.765	0.697
10^2	10^3	0.052	0.322	0.267	0.313
10^3	10^4	0.019	0.039	0.058	0.064
10^4	10^5	0.007	0.011	0.024	0.009
10^5	10^6	0.088	0.081	0.080	0.079
10^6	10^7	2.144	2.168	2.131	2.144

Table 4.30

The value of ΔI_γ for ^{236}U , in barn.

$E_{\min}$	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
(eV)					
0.5	10	749.7	274.5	283.8	265.7
10	10^2	43.59	46.04	35.67	40.28
10^2	10^3	31.06	19.67	17.03	19.60
10^3	10^4	4.480	3.820	3.660	4.760
10^4	10^5	1.160	0.950	1.046	1.690
10^5	10^6	1.060	0.770	0.154	0.130
10^6	10^7	0.210	0.120	0.100	0.090

Table 4.31

The value of $\langle \sigma_f \rangle$ for ^{236}U , in barn.

$E_{\min}$	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
(eV)					
0.5	10	0.500	1.851	1.915	1.806
10	10^2	0.269	0.442	0.448	0.404
10^2	10^3	0.017	0.104	0.097	0.101
10^3	10^4	0.008	0.012	0.021	0.020
10^4	10^5	0.003	0.003	0.007	0.003
10^5	10^6	0.078	0.076	0.076	0.071
10^6	10^7	1.125	1.125	1.103	1.113

Table 4.32

The value of $\langle \sigma_{\gamma} \rangle$ for ^{236}U , in barn.

$E_{\min}$	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
(eV)					
0.5	10	434.9	156.4	161.8	151.6
10	10^2	26.28	27.38	21.90	25.05
10^2	10^3	9.912	6.399	5.573	6.278
10^3	10^4	1.568	1.310	1.317	1.606
10^4	10^5	0.565	0.444	0.489	0.486
10^5	10^6	0.285	0.190	0.174	0.173
10^6	10^7	0.048	0.022	0.022	0.019

4.6 ^{237}U .

Even-odd ^{237}U nucleus is a short lived one ($T_{1/2}=6.75$ days) transformed by β^- decay to a practically stable ^{237}Np nucleus. ^{237}U generation in active zones and in reactor blankets is due to the $^{238}\text{U}(n,2n)$ and $^{235}\text{U}(n,\gamma)$ $^{236}\text{U}(n,\gamma)$ reaction. ^{237}U production is one of two connection channels of uranium chain with neptunium one and then with plutonium (^{239}U production under neutron capture of ^{238}U nucleus is the second channel). ^{237}U transmutation in thermal and resonance neutron flux is due to neutron capture and in neutron flux of fission spectrum and thermonuclear neutrons it occurs by fission way. Among the versions of evaluated neutron data compared by us the evaluations of characteristics of interaction of neutrons with ^{237}U nucleus are given in ENDL-82 and ENDF/B-6 only.

R.Howerton (LNLL,USA) is the author of ENDL-82 (MAT 7870)

evaluation; 1983 is a year of evaluation completion.

ENDF/B-6 (MAT 9234) evaluation on cross-sections agree, except format of data representation, with ENDF/B-5 evaluation assembled by Kinsey (BNL,USA). 1976 is a year of evaluation completion. ENDF/B-5 is a combination of evaluations of Benjamin and MacCrosson (SRL) and R. Howerton (LNLL). The region of resolved resonances is between 0 and 102.5 eV. Resonance parameters were determined by GENRPAR code [Mc71] with the data from SRL using the 3.5 eV average level spacing [Mc74]. The 2.109 eV level at negative energy presented in data from SRL evaluations was removed. Its contribution was changed by corresponding non-resonance cross-sections at the energy dependence of total, fission and capture cross-sections. The region of unresolved resonances is limited by the energy 10 keV from above. Average resonance parameters are based on the same data as parameters of resolved levels. Above 10 keV cross-section evaluations correspond to results from [Ho76].

In Tables 4.33-4.37 the comparison of evaluations of some characteristics of interaction between neutrons and ^{237}U nucleus are given.

Table 4.33

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{237}U , in barn.

Value	ENDL-82	ENDF/B-6	Recommend. value	Year of public.
σ_{f0}	0.350	2.000	2.000	1970, [2]
			<0.35	1984, [6]
$\sigma_{\gamma0}$	380.3	476.0	411_{-138}^{+}	1973, [5]
			443_{-167}^{+}	1984, [6]

Table 4.34

The values of I_f , I_γ for ^{237}U , in barn.

Value	ENDL-82	ENDF/B-6	Recommend. value	Year of public.
I_f	94.5	8.94	0	1970, [2]
I_γ	373.5	312.1	290	1973, [5]
			1200_{-200}^{+}	1984, [6]

Table 4.35

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{237}U , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	0.656	0.103	0.033	1.491	0.321	0.779
ENDF/B-6	0.699	0.080	0.033	1.491	0.321	0.779

Table 4.36

The values of ΔI_f , ΔI_γ for ^{237}U , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ΔI_f	ΔI_γ		
	ENDL-82	ENDF/B-6	ENDL-82	ENDF/B-6
0.5 10	44.02	0.835	240.5	118.6
10 10^2	31.82	1.550	95.23	154.2
10^2 10^3	9.744	0.419	26.97	31.93
10^3 10^4	3.278	1.362	7.390	5.970
10^4 10^5	2.039	1.474	2.900	1.300
10^5 10^6	1.989	1.487	0.380	0.040
10^6 10^7	1.568	1.814	0.090	0.090

Table 4.37

The values of $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{237}U , in barn.

$E_{\min}$ $E_{\max}$ (eV)	$\langle \sigma_f \rangle$	$\langle \sigma_\gamma \rangle$		
	ENDL-82	ENDF/B-6	ENDL-82	ENDF/B-6
0.5 10	21.87	0.516	76.19	55.94
10 10^2	11.12	0.532	32.95	52.83
10^2 10^3	3.404	0.172	9.197	10.46
10^3 10^4	1.222	0.820	2.583	1.754
10^4 10^5	0.874	0.609	0.942	0.346
10^5 10^6	0.856	0.679	0.217	0.143
10^6 10^7	0.814	0.891	0.022	0.023

4.7 ^{238}U .

Even-even ^{238}U nucleus is practically stable one ($T_{1/2}=4.468\cdot 10^9$ years, α - decay) of nature origin (99.275% in nature uranium nuclei mixture). It is a main fuel component of modern power reactors (about 90% of uranium content) and initial material for ^{239}Pu production in U-Pu fuel cycle due to the $^{238}\text{U}(n,\gamma)^{239}\text{U}(\beta^-)^{239}\text{Np}(\beta^-)^{239}\text{Pu}$ reaction chain. Neutron capture by ^{238}U is a main process of ^{238}U destruction in thermal and resonance neutron flux. Fission reaction is a main process in fission neutron flux, and in thermonuclear ones the $(n,2n)$ reaction, leading to ^{237}Np generation by the $^{238}\text{U}(n,2n)^{237}\text{U}(\beta^-)^{237}\text{Np}$ reaction chain, may compete with this process. One can ignore possible regeneration of ^{238}U in the $^{237}\text{U}(n,\gamma)$ or $^{239}\text{U}(n,2n)$ reactions because of small half-life of target nuclei (6.75 days for ^{237}U and 23.5 min for ^{239}U).

Characteristics of interaction of neutrons with ^{238}U have been, perhaps, studied more extensively and for a more extended time in comparison with other actinides, that is due to accessibility of material under investigation and of necessity to use them in power reactors with extended fuel reproduction. Evaluation of these characteristics are represented in all compared by us versions of evaluated neutron data.

ENDL-82 (MAT 7871) evaluation was carried out by R.Howerton and was completed in 1983.

BROND-2 (MAT 9271) evaluation was carried out by M.Nikolaev, L.Abagyan, N.Bazazyants and et al. (FEI, Russia) in 1978. In 1981 the revision was performed and in 1985 the checking was made. Resonance

parameters in the region of resolved resonance are conventionally presented as a mixture of two "isotopes". First "isotope" contains data for s and d neutrons, the second one contains data for p - neutrons. Thanks to this, it has been made possible to introduce various limits of resolved and unresolved resonance regions and various radii of potential scattering for s- and p-waves. Fission width is assumed to be zero, but nonzero subthreshold fission cross-section was added to the file of smooth cross-sections. The region of resolved s and d- resonances is between the limits 1 eV and 4650 eV and the corresponding file contains parameters of 249 s-resonances. Radiative widths were reduced for the first five resonances according to the data from [Bl77]. Multilevel Breit-Wigner formula is recommended for cross-section calculations in this range. In the region of unresolved s-and d- resonances (4.65 keV-200 keV) average resonance parameters at 34 energy points are given. They were reevaluated with regard to the data from [Ma83]. The region of resolved p-resonances is between the limits 1 eV and 2150 eV. Relevant file contains parameters of 232 resonances. Single-level Breit-Wigner formula is recommended for cross-section calculations. The region of unresolved p-resonances is between the limits of 2.25 keV and 200 keV. Average resonance parameters at 39 energy points are given. In the interval from 10^{-5} eV to 1 eV total, elastic scattering and capture cross-sections calculated by resolved resonances parameters are given point by point. In the interval from 1 eV to 200 keV smooth background to resonance subthreshold fission cross-sections are assumed. Cross-section evaluations at $E_n > 200$ keV were carried out in [Ni79].

ENDF/B-6 evaluation was completed in 1989 and it was carried out

by L.Weston, P.Young and W.Poenitz (ORNL, LANL, USA). Data on elastic and inelastic scattering were taken from ENDF/B-5 evaluation. The region of resolved resonances is limited by the energy 10 keV from above. Cross-sections in thermal point are determined by parameters of resolved resonances, thereat recommendations of Standards Committee on capture cross-sections were accounted. Resonance parameter evaluations are based on the results of D.Olsen (ORNL), R.Macklin (ORNL), M.Moxon (Harwell), M.Sowerby (Harwell). The region of unresolved resonances is limited by the energy 149 keV from above. Average resonance parameters were evaluated by F. Froehner (KFK) with the use of FITACS code (fitting to experimental data). The region of unresolved resonances is to be used only for self-shielding calculations and file 3 should be used to determine dilute cross-sections. Above the region of unresolved resonances capture cross-section was obtained immediately after standard analysis (of experimental data), performed in [Ca85] and [Po89]. Fission cross-section evaluation at this range was carried out by the same authors. The (n,2n) reaction cross-section evaluation was carried out by covariant analysis of available data from [Ba66], [Fr80], [Ko80], [Ch78], [Ma69], [Ma72], [Ka78], [La73], [Ro57], [Ph56], [Pe61], [Ve78], [An58], [Ry80]. The GNASH analysis has given the error of evaluation about 20%.

JENDL-3 (MAT 3926) evaluation was carried out in 1987 by Y. Kanda with co-authors (JAERI, Japan). The region of resolved resonances is between the limits 10^{-5} eV and 9.5 keV. Multilevel Breit-Wigner formula is used. Resonance parameters at 6.67 eV level were fitted for reproduction of recommended at cross-sections thermal point. Resonance

energies and neutron width were taken weighted average of data from [Ol84] and JENDL-2 evaluation. Radiative and fission widths were taken from JENDL-2 and effective scattering radius was assumed to be equal to 9.7 Fermi. The region of unresolved resonances is limited by the energy 50 keV from above. Average resonance parameters were obtained by ASREP (Y.Kikuchi) fitting code with the aim of reproduction of known at this range cross-sections. In the region of resolved non-resonance cross-sections given in smooth cross-section file were assumed. At the energy above 50 keV in this file the $(n,2n)$ reaction cross-section up to 15 MeV was evaluated by data from [Fr80], and above 15 MeV it was done by data from [Ve78] and [Ka79]. The $(n,3n)$ reaction cross-section evaluation is based on data from [Ve79]. From 50 keV to 100 keV fission cross-section evaluation was carried out according to data from [Di77], for the interval from 100 keV to 600 keV data from [Di77], [Be77], [No76], [Me72], [Me75] were used; above 600 keV the results of evaluation from [Me72] corrected in terms of experimental data obtained later 1972 were used. From 50 keV to 300 keV capture cross-section evaluation was based mainly on measurement results from [Ka86]. Above 300 keV JENDL-2 evaluation verified by measurements of [Po75], [Pa75], [Mo71], [Fr70], [Me68] and slightly changed at high energies was used.

In Tables 4.38-4.44 calculated characteristics of interaction of neutrons with ^{238}U neutrons obtained by ENDL-82, JENDL-3, ENDF/B-6 and BROND-2 evaluations are compared among themselves and with recommended data are given.

Table 4.38

- The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{238}U , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend& value	Year of public.
σ_{f0}	0	$1.1 \cdot 10^{-7}$	$4.8 \cdot 10^{-5}$	0	$<5 \cdot 10^{-4}$ $4 \cdot 10^{-6} \text{ *)}$	1973, [5] 1984, [6]
$\sigma_{\gamma0}$	2.736	2.681	2.711	2.709	$2.70^{+0.02}_{-0.02}$ $2.68^{+0.02}_{-0.02}$	1973, [5] 1984, [6]

*) - calculated value

Table 4.39

The values of I_f , I_γ for ^{238}U , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	1.272	1.250 (2&020) *)	1.242	1.251 ($7.8 \cdot 10^{-6} \text{ **})$)	-	
I_γ	273.7	279.0 (279.0) *)	267.2	289.9 (276.3) **)	275^{+5}_{-5} 277^{+3}_{-3}	1973, [5] 1984, [6]

*) - $E_{\max} = 20 \text{ MeV}$

**) - $E_{\min} = 100 \text{ keV}$

Table 4.40

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{238}U , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	0.323	0.075	0.027	1.129	0.897	0.434
JENDL-3	0.318	0.065	0.018	1.136	0.910	0.350
ENDF/B-6	0.314	0.073	0.021	1.130	0.871	0.450
BROND-2	0.317	0.068	0.022	1.155	0.920	0.479

Table 4.41

The value of ΔI_f for ^{238}U , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	0	$6.4 \cdot 10^{-5}$	$6.3 \cdot 10^{-5}$	0
10	10^2	0	$2.2 \cdot 10^{-4}$	$2.2 \cdot 10^{-4}$	0
10^2	10^3	0	0.001	0.001	0
10^3	10^4	0	$4.2 \cdot 10^{-4}$	$3.7 \cdot 10^{-4}$	0
10^4	10^5	0	$3.1 \cdot 10^{-4}$	$1.6 \cdot 10^{-4}$	$1.6 \cdot 10^{-5}$
10^5	10^6	0.004	0.004	0.004	0.004
10^6	10^7	1.268	1.244	1.210	1.247

Table 4.42

The value of ΔI_{γ} for ^{238}U , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	130.9	131.1	111.0	137.4
10 10^2	116.4	122.2	128.5	125.3
10^2 10^3	21.87	21.34	22.05	22.61
10^3 10^4	3.080	2.948	3.020	3.210
10^4 10^5	0.730	0.941	2.480	0.996
10^5 10^6	0.630	0.266	0.110	0.304
10^6 10^7	0.080	0.067	0.080	0.080

Table 4.43

The value of $f < \sigma >$ for ^{238}U , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	0	$4.2 \cdot 10^{-5}$	$3.7 \cdot 10^{-5}$	0
10 10^2	0	$7.0 \cdot 10^{-5}$	$7.0 \cdot 10^{-5}$	0
10^2 10^3	0	$8.7 \cdot 10^{-4}$	$8.7 \cdot 10^{-4}$	0
10^3 10^4	0	$9.4 \cdot 10^{-5}$	$8.4 \cdot 10^{-5}$	0
10^4 10^5	0	$9.8 \cdot 10^{-5}$	$6.5 \cdot 10^{-5}$	$1.3 \cdot 10^{-5}$
10^5 10^6	0.003	0.003	0.003	0.003
10^6 10^7	0.708	0.692	0.691	0.697

Table 4.45

The value of $\bar{\sigma}_f$ for ^{238}U , in barn.

ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Measurem. results	Year of public.
0.323	0.318	0.314	0.317	$0.320^{+0.009}_{-0.009}$	1975, [23]
				$0.319^{+0.008}_{-0.008}$	1976, [15]
				$0.317^{+0.006*}_{-0.006*}$	1981, [17]
				$0.320^{+0.006}_{-0.006}$	1982, [19]
				$0.326^{+0.007}_{-0.007}$	1984, [11]
				$0.332^{+0.005}_{-0.005}$	1985, [20]

*) - recommended value

Table 4.44

The value of ΔI_γ for ^{238}U , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	91.28	96.21	77.35	95.81
10	10^2	41.04	44.28	44.55	43.71
10^2	10^3	5.525	5.687	5.650	5.712
10^3	10^4	1.049	1.042	1.002	1.109
10^4	10^5	0.360	0.332	0.695	0.334
10^5	10^6	0.130	0.122	0.129	0.121
10^6	10^7	0.019	0.015	0.016	0.018

The cross-section of ^{238}U neutron fission reaction at overthreshold range is used as a standard under definition of fast neutron flux at relative fission cross-section measurements and in solution of the problem of high level dosimetry. The accuracy of knowledge of ^{238}U fission cross-section, has been reached to the present moment, satisfies to practical demands; but from the point of view of its using as a standard the refinements near 2 MeV are needed. The refinements are also necessary at threshold range ($E_n = 0.5$ MeV - 1.4 MeV) where according to ENDF/B-5 (MAT 6938) evaluation the error of knowledge of ^{238}U fission cross-section noticeably exceeds the desired level 1% and accounts for about 7-10%. Table 4.45 is given for illustration where σ_f calculated values for ^{238}U are compared with various years published measurement results.

4.8 ^{239}U .

Even-odd ^{239}U nucleus is a short-lived one ($T_{1/2}=23.3$ min) transforming to ^{239}Np due to β - decay. The last, in its turn, with $T_{1/2}=2.355$ days, transforms into ^{239}Pu nucleus due to β - decay. Small life-time of ^{239}U makes hardly probable the competition of neutron-nuclear reaction to β - decay under reactor conditions. Thus in order to provide the competition of ^{239}U neutron fission process with β - decay it is necessary to reach the density of thermal neutron flux about $1.4 \cdot 10^{19} \text{ cm}^{-2} \text{ sec}^{-1}$. Such value under reactor conditions was unattainable. It is unlikely on this reason that definition of characteristics of interaction between neutrons and ^{239}U nucleus will be required with the same accuracy that is demanded to the knowledge of characteristics, for example, of ^{238}U . The $^{238}\text{U}(n,\gamma)^{239}\text{U}(\beta)^{239}\text{Pu}$ reaction chain may be changed without great uncertainties for direct transition from ^{238}U to ^{239}Pu .

In compared by us version of evaluated neutron data the evaluations of characteristics of interaction of neutrons with ^{239}U are listed only in ENDL-82 (MAT 7872) [Mo83]. The evaluations of such characteristics were contained in ENDF/B-5 (MAT 8239) evaluation also, but these data have not been transferred to ENDF/B-6 evaluation, therefore we shall not discuss them.

In Tables 4.46-4.47 some characteristics of interaction of neutrons with ^{239}U nucleus calculated by ENDL-82 and recommended values presented in literature are given.

Table 4.46

The values of σ_{f0} , $\sigma_{\gamma 0}$, I_f , I_γ , $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{239}U , in barn.

Value	ENDL-82	recommend. value	Year of public.	Value	ENDL-82
σ_{f0}	14.02	14^{+3}_{-3}	1973, [5]	I_f	263.1
		14^{+3}_{-3}	1984, [6]	I_γ	156.2
				$\bar{\sigma}_f$	0.528
$\sigma_{\gamma 0}$	22.02	22^{+5}_{-5}	1973, [5]	$\bar{\sigma}_\gamma$	0.056
		22^{+5}_{-5}	1984, [6]	$\bar{\sigma}_{2n}$	0.086
σ_f	0.969			σ_{2n}	0.250
				σ_{3n}	1.400

Table 4.47

The values of ΔI_f , ΔI_γ , $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{239}U , in barn.

ENDL-82 evaluation					
$E_{\min}$	$E_{\max}$	ΔI_f	ΔI_γ	$\langle \sigma_f \rangle$	$\langle \sigma_\gamma \rangle$
(eV)					
0.5	10	146.8	92.97	66.02	39.03
10	10^2	78.27	44.01	27.39	15.28
10^2	10^3	24.00	12.86	8.375	4.436
10^3	10^4	7.560	4.030	2.708	1.480
10^4	10^5	3.380	2.140	1.356	0.706
10^5	10^6	1.920	0.190	0.680	0.122
10^6	10^7	1.150	0.050	0.558	0.011

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5. Isotopes of ^{237}Np .

5.1. ^{237}Np -

Odd-even ^{237}Np nucleus is a long-lived one ($T_{1/2}=2.14 \cdot 10^6$ years, α -decay). The $^{235}\text{U}(n,\gamma)$ $^{236}\text{U}(n,\gamma)$ $^{237}\text{U}(\beta^-)$ ^{237}Np and $^{238}\text{U}(n,2n)$ $^{237}\text{U}(\beta^-)$ ^{237}Np reaction chains are the generation channels in nuclear reactors. ^{237}Np nucleus is used as an original material for obtaining of ^{238}Pu nuclei employed in radioisotope equipment in the capacity of α -radiation source in neutron sources (in combination with Be) and also as radioisotope energy source. ^{238}Pu production is due to $^{237}\text{Np}(n,\gamma)$ $^{238}\text{Np}(\beta^-)$ ^{238}Pu reaction chain. ^{238}Np nucleus is noticeably broken down only by the action of fast neutrons (effective "threshold" of fission reaction is about 0.5 MeV), thereat in thermal and resonance neutron spectra radiation capture is a main ^{237}Np transmutation process. In fission and thermonuclear neutron spectra fission is a main process of ^{237}Np destruction. The $^{235}\text{U}(n,\gamma)$ $^{236}\text{U}(n,\gamma)$ $^{237}\text{U}(\beta^-)$ $^{237}\text{Np}(n,\gamma)$ $^{238}\text{Np}(\beta^-)$ $^{238}\text{Pu}(\alpha)$ $^{234}\text{U}(n,\gamma)$ ^{235}U closed reaction chain takes place owing to transmutation sequence; although, because of relatively great value of $T_{1/2}=86.4$ years for ^{238}Pu , processes of interaction of neutrons with this nucleus leading either to ^{239}Pu or to fission fragment production are more probable.

In compared by us versions of evaluated neutron data the evaluations of characteristics of interaction between neutrons and ^{237}Np nucleus are given in ENDL-82, JENDL-3 and ENDF/B-6. Data evaluation for ^{237}Np was presented in the first BROND (BROND-1) version, but these data were not conveyed to later BROND-2 one, therefore they are not discussed there.

R. Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7874)

evaluation; 1983 is a year of evaluation completion.

P. Young, E. Arthur, F. Mann (LANL, USA) are the authors of ENDF/B-6 (MAT 9346) evaluation; 1990 is a year of evaluation completion. The region of resolved resonances is between the limits 10^{-5} eV and 150 eV. Data from [De80] and [De89] were used for evaluation. In the range of smooth cross-sections at $E_n > 150$ eV the (n,2n), (n,3n) and (n,4n) reaction cross-sections were calculated by GNASH code [Ar88], [Yo77]. Thereat for the (n,2n) reaction experimental data from [Ko85] were used in the interval from 9 MeV and below. Fission cross-section evaluation up to 0.8 MeV is based on the results of [De80] and [De89]. At $0.9 \text{ MeV} \leq E_n \leq 20 \text{ MeV}$ empirical fit mainly to experimental data from [Li88], [Me83] and [Me88] was used together with the data from [Te86], [Ka85], [Ca82], [Wh67], [Za84], [Va82]. Capture cross-section up to 5 MeV was evaluated on data from [De80], [De89] with agreement with experimental data from [We81]. Above 5 MeV ENDF/B-5 evaluation is retained.

Y. Uenohara, Y. Kanda (Kyushu Univ., Japan) are the authors of JENDL-3 (MAT 3931) evaluation. 1987 is a year of evaluation completion. The region of resolved resonances is between the limits 10^{-5} and 130 eV. Single-level Breit-Wigner formula is used for cross-section calculations. Resonance energy evaluations and neutron and radiative widths are based on results of [We81], and those of fission widths are based on results from [Pl76]. Average radiative width is equal to 40 meV. Resonance at negative energy was adopted. The region of unresolved resonances is limited by the energy 30 keV from above. Evaluations from [We81] were used with some changes. Following values of averaged parameters were used: $S_0 = 1.02 \cdot 10^{-4}$, $S_1 = 1.888 \cdot 10^{-4}$, $\bar{D} = 0.45$ eV, $\bar{\Gamma}_\gamma = 40$ meV. Average fission width was determined so that in this region

$\sigma_f = 0.009$ barn. At $E_n > 30$ keV CASTHY code [Ig75] was used for capture cross-section calculation. Level scheme corresponded to compilation of [E171]. Level continuum was introduced above 0.370 MeV. Level density parameters corresponded to data from [Gi65]. The strength function for γ -radiation was the same that at $E_n = 200$ keV σ_γ was equal to 0.742 barn. In evaluation of the (n,2n) reaction cross-section data from JENDL-2, calculated by evaporation model [Se78] were fitted to experimental data from [Pe61], [La69], [Li75], [Fu82], [Gr83], [Ko83]. Above 16.5 MeV the (n,3n) reaction cross-sections from JENDL-2, obtained by the same model as for the (n,2n) reaction cross-sections, were changed by addition of difference between (n,2n) reaction cross-section from JENDL-2 and the same one from JENDL-3. Below 16.5 MeV the (n,3n) reaction cross-section was renormalized so as to agree with JENDL-3 evaluation at 16.5 MeV. At fission cross-section evaluation above 100 keV there were used experimental data from [K147], [Pr58], [Sc59], [Gr67], [Iy69], [Ji72], [Ko73], [Ar81], [Ca82], [Ga83], [Ku78], [Wh65], [Wh67], [St68], [Be82] and [Me83].

In Tables 5.1-5.5 calculated values of some characteristics of interaction of neutrons with ^{237}Np obtained by evaluated data from ENDL-82, JENDL-3 and ENDF/B-6 are compared among themselves and with recommended data.

Table 5.3

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{237}Np , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.305	0.166	0.007	2.053	0.500	0.500
JENDL-3	1.340	0.167	0.005	2.247	0.493	0.088
ENDF/B-6	1.329	0.183	0.005	2.111	0.482	0.012
Measurement	$1.332^{+0.038}_{-0.038}$, [23]					
value	$1.342^{+0.027}_{-0.027}$, ^{*)} [17]					
	$1.339^{+0.027}_{-0.027}$, [19]					
	$1.366^{+0.027}_{-0.027}$, [11]					

*) - recommended value

Table 5.4

The values of ΔI_f , ΔI_γ for ^{237}Np , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ΔI_f			ΔI_γ		
		ENDL-82	JENDL-3	ENDF/B-6	ENDL-82	JENDL-3	ENDF/B-6
0.5	10	0.044	0.025	0.020	351.9	418.4	409.0
10	10^2	0.816	0.099	0.093	170.0	170.8	178.5
10^2	10^3	0.420	0.028	0.053	48.59	52.02	48.43
10^3	10^4	0.131	0.021	0.023	14.24	15.09	13.89
10^4	10^5	0.029	0.027	0.025	5.340	4.894	5.200
10^5	10^6	0.825	0.837	0.827	1.580	1.718	1.080
10^6	10^7	3.806	3.904	3.901	0.090	0.090	0.110

Table 5.1

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{237}Np , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	Recommend. value	year of public.
σ_{f0}	0.015	0.019	0.018	$0.019^{+0.003}_{-0.003}$	1973, [5]
				$0.0215^{+0.0024}_{-0.0024}$	1984, [6]
$\sigma_{\gamma0}$	169.7	181.0	181.0	169^{+3}_{-3}	1973, [5]
				$175.9^{+2.9}_{-2.9}$	1984, [6]

Table 5.2

The values of I_f , I_γ for ^{237}Np , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
I_f	6.070	4.941	4.941	6.01	1973, [24]
		$(6.360)^{*})$		$6.9^{+1.0}_{-1.0}$	1984, [6]
I_γ	591.7	663.0	656.2	660^{+50}_{-50}	1973, [5]
		$(663.0)^{*})$		640^{+50}_{-50}	1984, [6]

*) - $E_{\text{max}} = 20 \text{ MeV}$

Table 5.5

The values of $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{237}Np , in barn .

$E_{\min}$ $E_{\max}$ (eV)	$\langle \sigma_f \rangle$			$\langle \sigma_\gamma \rangle$		
	ENDL-82	JENDL-3	ENDF/B-6	ENDL-82	JENDL-3	ENDF/B-6
0.5 10	0.019	0.007	0.007	91.51	102.9	101.8
10 10^2	0.400	0.042	0.039	58.13	61.79	64.40
10^2 10^3	0.138	0.010	0.019	16.39	18.05	16.85
10^3 10^4	0.038	0.009	0.009	5.282	5.247	4.805
10^4 10^5	0.014	0.014	0.012	1.962	1.922	1.765
10^5 10^6	0.641	0.649	0.641	0.417	0.403	0.453
10^6 10^7	1.762	1.799	1.815	0.021	0.018	0.026

5.2 ^{238}Np

Odd-odd ^{238}Np is a short-lived nucleus ($T_{1/2} = 2.117$ days) transforming into ^{238}Pu due to β^- -decay. The $^{235}\text{U}(n, \gamma)$ $^{236}\text{U}(n, \gamma)$ $^{237}\text{U}(\beta^-)$ $^{237}\text{Np}(n, \gamma)$ ^{238}Np and $^{238}\text{U}(n, 2n)$ $^{237}\text{U}(\beta^-)$ $^{237}\text{Np}(n, \gamma)$ ^{238}Np reaction chains are the production channels in reactors. The ^{238}Np nucleus fission is a main transmutation process going under the action of thermal and resonance neutrons. This reaction is important in evaluations of ^{238}Np accumulation rate in the $^{237}\text{Np}(n, \gamma)$ $^{238}\text{Np}(\beta^-)$ ^{238}Pu reaction chain. The rate of ^{238}Np β^- -decay is compared with that of nucleus destruction in fission reactions at the densities of thermal neutron flux of $\sim 2 \cdot 10^{15} \text{ cm}^{-2} \text{ sec}^{-1}$; this value is achievable in reactors of special types. In thermonuclear neutron flux the $(n, 2n)$ reactions may compete with fission process. For competition with ^{238}Np

β^- - decay process the $\sim 10^{18} \text{ cm}^{-2} \text{ sec}^{-1}$ densities of neutron flux are required. Because of small $T_{1/2}$ value for ^{238}Np the characteristics of interaction of neutrons with this nucleus have been studied noticeably worse than those for, as an example, ^{237}Np nucleus. No measurements were carried out practically for capture resonance integral. Although some attempts to measure I_f values were made, the lower limit there has not been improved. Fission cross-section evaluation at thermal point is more solid. Among the versions of evaluated neutron data compared by us characteristics of interaction of neutrons with ^{238}Np nucleus are represented fully enough in ENDL-82 only.

R Howerton (LNLL, USA) is the author of ENDL-82 (MAT 8309) evaluation; 1983 is a year of evaluation completion.

ENDF/B-6 (MAT 9349) evaluation on cross-sections differs from ENDF/B-5 (MAT 8338) evaluation only by format of data representation. The last evaluation was carried out in SRL (USA). R. Benjamin and F. MacCrosson are the authors of this evaluation; 1975 is a year of evaluation completion. However in files MAT 9349 and MAT 8338 being at our disposal, in smooth cross-section files (at the energy above 10 keV) only elastic scattering cross-section was nonzero and equal to potential fission cross-section at the energy 10 keV, calculated by spherical optical model. Energy range below 10 keV was divided into two parts. The first is the resolved resonance region included between the limits 0 eV and 100.2 eV. Resonance parameters were determined by GENRBAR code from [Ma71] for the average 1.05 eV level spacing. Above 100.2 eV up to 10 keV the region of unresolved resonances lies. Data base from SRL (USA) with average resonance parameters was used for evaluation.

In Tables 5.6-5.10 some calculated characteristics of interaction

of neutrons with ^{238}Np nucleus obtained by ENDL-82 and ENDF/B-6 are compared among themselves and with recommended data.

Table 5.6

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{238}Np , in barn.

Value	ENDL-82	ENDF/B-6	Recommend. value	Year of public.
σ_{f0}	2071	2031	2070^{+30}_{-30}	1973, [5]
			2088^{+30}_{-30}	1984, [6]
$\sigma_{\gamma0}$	100.0	203.2	43	1970, [21]
			43	1973, [24]
$\sigma_{f0} + \sigma_{\gamma0}$	2171	2234	2243	1970, [21]

Table 5.7

The values of I_f , I_γ for ^{238}Np , in barn.

Value	ENDL-82	ENDF/B-6	Recommend. value	Year of public.
I_f	1390	895.7	880	1972, [25]
			1441	1973, [24]
			883^{+70}_{-70}	1984, [6]
I_γ	56.79	100.0	600	1972, [25]
$I_f + I_\gamma$	1447	995.7	1480	1972, [25]
			1500^{+500}_{-500}	1984, [6]

*) - $E_{\text{max}} = 10 \text{ MeV}$

Table 5.8

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{238}Np , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-3	1.473	0.033	0.021	1.850	0.580	0.500

Table 5.9

The values of ΔI_f , ΔI_γ for ^{238}Np , in barn.

$E_{\min}$ $E_{\max}$ (eV)		ΔI_f		ΔI_γ	
		ENDL-82	ENDF/B-6	ENDL-82	ENDF/B-6
0.5	10	998.2	696.8	42.27	74.58
10	10^2	243.5	141.7	9.448	15.73
10^2	10^3	92.10	43.42	3.310	7.640
10^3	10^4	34.70	13.87	1.164	2.063
10^4	10^5	11.50	0	0.410	0
10^5	10^6	6.600	0	0.141	0
10^6	10^7	3.400	0	0.050	0

Table 5.10

The values of $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{238}Np , in barn.

$E_{\min}$ $E_{\max}$ (eV)		$\langle \sigma_f \rangle$		$\langle \sigma_\gamma \rangle$	
		ENDL-82	ENDF/B-6	ENDL-82	ENDF/B-6
0.5	10	248.5	177.5	10.27	19.35
10	10^2	88.52	49.55	3.384	5.503
10^2	10^3	42.29	15.13	1.186	2.647
10^3	10^4	12.63	4.799	0.417	0.632
10^4	10^5	4.766	0	0.146	0
10^5	10^6	1.816	0	0.051	0
10^6	10^7	1.523	0	0.018	0

5.3 ^{239}Np .

Odd-even ^{239}Np nucleus is a short-lived one ($T_{1/2}=2.355$ days) transforming due to β^- -decay into ^{239}Pu . The $^{238}\text{U}(n,\gamma)^{239}\text{U}(\beta^-)^{239}\text{Np}$ reaction chain is a main generation channel in reactors. The $^{235}\text{U}(n,\gamma)^{236}\text{U}(n,\gamma)^{237}\text{U}(\beta^-)^{237}\text{Np}(n,\gamma)^{238}\text{Np}(n,\gamma)^{239}\text{Np}$ and $^{238}\text{U}(n,2n)^{237}\text{U}(\beta^-)^{237}\text{Np}(n,\gamma)^{238}\text{Np}(n,\gamma)^{239}\text{Np}$ reaction chains are the side and less productive channels because of small $T_{1/2} (^{238}\text{Np})$ value. Owing to small $T_{1/2} (^{239}\text{Np})$ value the characteristics of interaction of neutrons with this nucleus have not been studied practically. In particular, no data about measured and even evaluated resonance integrals are presented in literature. Only a publication on rough cross-section evaluation at thermal point is given. Quality of neutron data used will hardly probable influence on production of ^{239}Pu generation rates. On evaluations the process of ^{239}Np neutron radiation capture may start to compete with that of β^- -decay at the density of thermal neutron flux about $7 \cdot 10^{16} \text{ cm}^{-2} \text{ sec}^{-1}$. Among the versions of evaluated neutron data compared by us the evaluations of characteristics of interaction between neutrons and ^{239}Np nucleus are given in JENDL-3 and ENDF/B-6 only.

JENDL-3 (MAT 3932) evaluation differs from JENDL-2 (MAT 2932) one only by format of data representation. JENDL-2 evaluation differs from JENDL-1 (JENDL-CG) by expansion of determination range up to 20 MeV and by revision of energy distribution evaluations of secondary neutrons from various reactions. This work was done by Y. Kauda (Kyushu Univ., Japan) in 1976. Resonance parameters are missing from cross-section descriptions. Below 4 eV fission cross-section is assumed to be equal to zero. The $1/V$ dependence is adopted for capture cross-section at this range; cross-section at thermal point was evaluated by

experimental data from [St59]. Above 4 eV fission cross-section was evaluated according to the data for ^{237}Np fission cross-section with renormalizing on neutron separation energy. The (n,2n) and (n,3n) reaction cross-sections were calculated by model from [Pe65].

At $4 \text{ eV} \leq E_n \leq 100 \text{ keV}$ capture cross-section was determined with the help of the following formula: $\sigma_\gamma(E) = 4.35 \cdot (E_n(\text{eV}))^{-1/2}$ barn. At $E_n > 100 \text{ keV}$ energy dependence of experimental data for ^{237}Np from [Na71] normalized on 1.4 barn cross-section at the point 100 keV, was used.

ENDF/B-6 (MAT 9352, R. Wright, ORNL, USA) evaluation differs from JENDL-2 by format of data representation and by the fact, that at $E_n < 4 \text{ eV}$ capture cross-section is calculated by formula $77 \cdot (0.0253)^{1/2} E_n(\text{eV})^{-1/2}$ barn so that the capture cross-section agrees with the value recommended in [2] at thermal point.

In Tables 5.11-5.14 some recommended and calculated characteristics of interaction of neutrons with ^{239}Np nucleus are given.

Table 5.11

The values of σ_{f0} , $\sigma_{\gamma0}$, I_f , I_γ for ^{239}Np , in barn.

Value	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
σ_{f0}	0	0	< 1	1973, [5]
$\sigma_{\gamma0}$	37.00	77.00	45	1973, [5]
			68^{+10}_{-10}	1984, [6]
I_f	5.422	5.422		
	(7.060)*)			
I_γ	445.0	457.4		
	(445.0)			

*) - $E_{\text{max}} = 20 \text{ MeV}$

Table 5.12

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{239}Np , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
JENDL-3	1.461	0.188	0.005	2.420	0.143	0.246
(ENDF/B-6)						

Table 5.13

The values of ΔI_f , ΔI_γ for ^{239}Np , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ΔI_f	ΔI_γ		
		JENDL-3	ENDF/B-6	JENDL-3	ENDF/B-6
0.5	10	0		171.0	182.7
10	10^2	0		188.4	
10^2	10^3	0		59.64	
10^3	10^4	0		18.84	
10^4	10^5	0.041		6.060	
10^5	10^6	1.467		1.650	
10^6	10^7	3.914		0.110	

Table 5.14

The values of $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{239}Np , in barn.

$E_{\min}$ (eV)	$E_{\max}$	$\langle \sigma_f \rangle$		$\langle \sigma_\gamma \rangle$	
		JENDL-3	ENDF/B-6	JENDL-3	ENDF/B-6
0.5	10	0		108.3	110.0
10	10^2	0		66.20	
10^2	10^3	0		20.97	
10^3	10^4	0		6.632	
10^4	10^5	0.029		2.127	
10^5	10^6	0.990		0.466	
10^6	10^7	1.768		0.024	

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6. Isotopes of $^{238-244}\text{Pu}$.

6.1. ^{238}Pu .

Even-even ^{238}Pu nucleus is relatively long-lived one with $T_{1/2}=87.74$ years. It transforms into ^{234}U due to α -decay. ^{238}Pu nucleus is used in radioisotope instruments, neutron sources (in combination with Be) and radioisotope energy sources. ^{237}Np is an initial material for ^{238}Pu production. The $^{237}\text{Np}(n,\gamma)^{238}\text{Np}(\beta^-)^{238}\text{Pu}$ reaction chain corresponds to production process. The $^{238}\text{Pu}(\alpha)^{234}\text{U}(n,\gamma)^{235}\text{U}(n,\gamma)^{236}\text{U}(n,\gamma)^{237}\text{U}(\beta^-)^{237}\text{Np}(n,\gamma)^{238}\text{Np}(\beta^-)^{238}\text{Pu}$ closed reaction chain takes place, although because of relatively great value of $T_{1/2}$ for ^{238}Pu , interaction processes of this nucleus with neutrons are more probable. The process of neutron radiation capture by ^{238}Pu nucleus leading to production of ^{239}Pu nucleus is a dominant process in thermal and resonance neutron flux. ^{238}Pu fission process is more probable in fission and thermonuclear neutron flux. The evaluations of characteristics of interaction of neutrons with ^{238}Pu are presented in all compared by us versions of evaluated neutron data.

R. Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7876) evaluation. 1983 is a year of evaluation completion.

A. Klepatskii, V. Maslov, Yu. Porodzinskii, E. Sukhovitskii (IJE, Belorussia) are the authors of BROND-2 (MAT 9411) evaluation; 1987 is a year of evaluation completion. The region of resolved resonance parameters is between the limits 10^{-5} eV and 500 eV. Resonance parameter evaluations are based on experimental data from [Yo67], [St73], [Bu82]. Single-level Breit-Wigner formula is

used in this region to describe resonance cross-section dependence. The region of unresolved resonances is limited by the energy 150 keV from above. Average resonance parameters were obtained by parameters of resolved levels (with $S_0=1.237 \cdot 10^{-4}$, $\bar{D}(l=0)=9.103$ eV, $\bar{\Gamma}_\gamma=34$ meV, $\bar{\Gamma}_f(l=0)=46$ meV) and with the help of calculations by coupled channels method ($S_1=2.12 \cdot 10^{-4}$ for the ground state, $S_0=1.24 \cdot 10^{-4}$ and $S_1=2.08 \cdot 10^{-4}$ for the first excited state, $R=9.36$ Fermi). Level spacing and fission width are the energy dependent parameters. Distribution of fission widths from [An75] was used. Experimental data from [Si73] describe capture cross-section evaluation. Above 150 keV fission and capture cross-section evaluations were carried out using experimental data from [Bu82], [Er68], [Er70], [Fo67], [Fo69], [Ba67], [Al85] and theoretical estimations from [Si73]. Capture, the (n,2n) and (n,3n) reaction cross-sections were calculated by the method of coupled channels, on statistical model using the level density from the work [An85] and exciton model.

T. Kawakita, T. Nakagawa (MAPI, JAERI, Japan) are the authors of JENDL-3 (MAT 3942) evaluation. 1989 is a year of evaluation completion. The region of resolved resonances is between the limits 10^{-5} eV and 500 eV. Multilevel Breit-Wigner formula is used. Parameters of resonances below 10 eV were evaluated on experimental data from [Yo67] and those for the region above 10 eV were evaluated on experimental data from [Si73]. Two resonances at negative energies and a resonance at the energy 2.9 eV were adopted for cross-section reproduction at thermal point recommended in [6]. Above 500 eV capture cross-section was calculated by optical statistical model with the help of CASTHY

code [Ig75]. Strength function of γ -radiation was evaluated by average 9.5 eV level spacing and 0.04 eV radiation width.

The $(n,2n)$ and $(n,3n)$ reaction cross-sections were calculated by the method described in [Pe65]. Fission cross-section was mainly evaluated on measurement results from [Bu82]. Experimental data from [Si73], [Ba67], [Fo67], [Fo70], [Er70], [Dr68], [Dr70], [Sh72], [Al88] were also used.

F. Mann, R. Schenter, Alter, Dunford (HEDL, AI, USA) are the authors of ENDF/B-6 (MAT 9434) evaluation; 1978 is a year of evaluation completion. Parameters of resolved resonances were taken from [Yo67]. Parameters in the region of unresolved resonances were evaluated by resolved ones and calculated by optical model (they preserved unchanged from ENDF/B-1 version). In smooth cross-section range the $(n,2n)$ and $(n,3n)$ reaction cross-sections were calculated by statistical model from [Ma77]. Fission cross-section evaluation was based on experimental data from [Er68], [Fo70], [Dr70], [Si73]. Capture cross-section was calculated by statistical model from [Ja70].

In Tables 6.1-6.7 characteristics of interaction between neutrons and ^{238}Pu nucleus, corresponding to ENDL-82, JENDL-3, ENDF/B-6 and BROND-2 evaluation are compared among themselves and with recommended data available.

Table 6.1

The values of σ_{f0} , $\sigma_{\gamma 0}$ for ^{238}Pu , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
σ_{f0}	16.31	17.89	17.01	17.20	$16.5^{+0.5}_{-0.5}$	1973, [5]
					$17.9^{+0.4}_{-0.4}$	1984, [6]
$\sigma_{\gamma 0}$	547.3	540.3	552.4	544.4	547^{+20}_{-20}	1973, [5]
					540^{+7}_{-7}	1984, [6]

Table 6.2

The value of I_f , I_γ for ^{238}Pu , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	30.50	30.82	28.94	37.09	24^{+4}_{-4}	1973, [5]
		(32.70) ^{*)}			33^{+5}_{-5}	1984, [6]
I_γ	152.1	153.9	155.7	150.5	141^{+15}_{-15}	1973, [5]
		(154.0) ^{*)}			162^{+15}_{-15}	1984, [6]

*) - $E_{\text{max}} = 20 \text{ MeV}$

Table 6.3

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{238}Pu , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	2.048	0.110	0.002	2.717	0.190	0.010
JENDL-3	2.000	0.097	0.006	2.722	0.515	0.054
ENDF/B-6	1.984	0.142	0.077	2.668	0.630	0.500
BROND-2	1.968	0.101	0.001	2.651	0.134	-

Table 6.4

The value of ΔI_f for ^{238}Pu , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	2.649	2.938	2.955	3.131
10 10^2	3.898	3.427	3.349	3.123
10^2 10^3	9.852	12.01	9.517	14.57
10^3 10^4	3.710	2.780	3.745	6.494
10^4 10^5	2.191	1.748	1.434	2.084
10^5 10^6	2.979	2.643	2.687	2.415
10^6 10^7	5.219	5.284	5.257	5.274

Table 6.5

The value of ΔI_γ for ^{238}Pu , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	54.72	58.99	60.07	57.96
10 10^2	54.93	51.14	48.69	49.36
10^2 10^3	36.23	34.71	35.92	34.77
10^3 10^4	4.650	6.440	8.010	6.160
10^4 10^5	1.240	1.720	2.310	1.750
10^5 10^6	0.200	0.320	0.530	0.380
10^6 10^7	0.150	0.090	0.150	0.100

Table 6.6

The value of $\langle \sigma_f \rangle$ for ^{238}Pu , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	1.491	1.638	1.661	1.701
10 10^2	1.773	1.451	1.546	1.331
10^2 10^3	3.383	4.036	3.330	6.259
10^3 10^4	1.444	1.098	1.497	2.111
10^4 10^5	0.925	0.686	0.633	0.800
10^5 10^6	1.556	1.485	1.447	1.370
10^6 10^7	2.297	2.409	2.397	2.408

Table 6.7

The value of $\langle \sigma_\gamma \rangle$ for ^{238}Pu , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	16.97	18.61	18.53	18.20
10 10^2	18.71	17.33	16.43	16.98
10^2 10^3	10.30	10.23	10.47	10.04
10^3 10^4	1.431	2.134	3.129	1.963
10^4 10^5	0.355	0.708	0.753	0.584
10^5 10^6	0.173	0.181	0.282	0.193
10^6 10^7	0.044	0.020	0.038	0.021

6.2. ^{239}Pu

Even-odd ^{239}Pu nucleus is a long-lived one ($T_{1/2}=2.4419 \cdot 10^4$ years, α -decay). Along with ^{238}U nucleus the ^{239}Pu one may become the main energy release component in fuel of future nuclear power

reactors. The $^{238}\text{U}(n,\gamma)^{239}\text{U}(\beta^-)^{239}\text{Np}(\beta^-)^{239}\text{Pu}$ reaction chain is a main ^{239}Pu generation channel in reactors-breeders. Following transmutation chains are the additional channels: $^{238}\text{U}(n,2n)^{237}\text{U}(\beta^-)^{237}\text{Np}(n,\gamma)^{238}\text{Np}(\beta^-)^{238}\text{Pu}(n,\gamma)^{239}\text{Pu}$ and $^{235}\text{U}(n,\gamma)^{236}\text{U}(n,\gamma)^{237}\text{U}(\beta^-)^{237}\text{Np}(n,\gamma)^{238}\text{Np}(\beta^-)^{238}\text{Pu}(n,\gamma)^{239}\text{Pu}$. ^{239}Pu nuclei undergo fission by neutrons of all energies. In thermal neutron flux fission of ^{239}Pu is a dominant process in that of ^{239}Pu depletion as well as in fission and thermonuclear neutron flux. In resonance neutron flux neutron capture process, leading to production of ^{240}Pu which disintegrates noticeably under the action of fast neutrons only, can compete with fission process.

Characteristics of interaction of neutrons with ^{239}Pu nucleus have been studied theoretically and experimentally for tens of years and are represented in all compared by us versions of evaluated neutron data.

R.Howerton (LNL, USA) is the author of ENDL-82 (MAT 7876) evaluation; 1983 is a year of evaluation completion.

V.Kon'shin, E. Sukhovitskii, G.Antsipov, A.Klepatskii, V.Maslov, G.Morogovskii and L.Bakhanovich (IJE, Belorussia) are the authors of BROND-2 (MAT 9439) evaluation; 1985 is a year of evaluation completion. The region of resolved resonances is between the limits 10^{-5}eV and 520 eV. Resonance parameters were obtained from the analysis of following experimental data: total cross-section from [Bo57], [Ig64], [De67]; fission cross-section from [De67], [GW71], [De72], [Br70]; capture cross-section from [GW71]. A satisfactory description of experimental data within the experimental errors has been reached only in the interval 10^{-5}eV - 1 eV providing the account of two resonances at negative

energies. However, fission cross-section at the range 0.7 - 1 eV. and fission and capture ones at 0.02 eV - 0.2 eV are calculated on resonance parameters with the errors 6% - 9% and 2% - 4% respectively. Therefore in these ranges the cited cross-sections in smooth cross-section file (MF 3) are given pointwise. In the region of unresolved resonances (0.5 keV - 100 keV) the average resonance parameters were obtained by a consistent analysis of data from the region of resolved resonances and by data on average cross-sections: total cross-section from [Ut64]; fission cross-section from [Ko81]; α -value from [Ko81]. The analysis was made in terms of neutron inelastic scattering, the $(n, \gamma f)$ reaction, level direct excitation and energy dependence of radiative width. Obtained average resonance parameters permit to reproduce fluctuations of average cross-sections in chosen energy intervals. At $E_n > 100$ keV fission cross-section and α value were evaluated in [Ko81] with regard to correlation of partial errors of various experiments. Fission cross-section was normalized to ENDF/B-5 evaluation of ^{235}U fission cross-section. The evaluated data for σ_f were obtained on the basis of experimental data from [Sc74]. In calculation of capture and the $(n, 2n)$, $(n, 3n)$ reaction cross-sections by statistical model coefficients of neutron transmission obtained by the method of coupled channels were used. In evaluation by the model of level density the collective effects (rotational and vibration) were accounted.

M. Kawai, K. Hida, T. Yoshida (NAIG, Japan) are the authors of JENDL-3 (MAT 3943) evaluation; 1990 is a year of evaluation completion. The region of resolved resonances is limited by the energy 1 keV from above. Formalism of Reich-Moore is used for

description of cross-section energy dependence. Level parameters were evaluated on the base of results from [Pe86] and [De87] where the result of fission cross-section measurements from [We84] were used as a basis for analysis carrying out.

Parameters given in [De87] were reevaluated in 1988, and final results came into present evaluation. The region of unresolved resonances is limited by the energy 30 keV from above. The energy dependence of S_0 , S_1 and fission width were determined so as to reproduce the evaluated values of total, capture and fission cross-sections. Below 1 keV there was adopted a background for fission cross-section to reproduce the results of identical measurements from [We84]. At $E_n > 30$ keV and below 1 MeV capture cross-section was evaluated by the product of evaluated fission cross-section and α . The α value corresponded to JENDL-2 evaluation. Above 1 MeV capture cross-section was calculated by code combination of CASTHY (optical-statistical model [Ig75]) and EGIS (method of coupled channels [Ra70]). Photon strength function at CASTHY code calculations was normalized so that at $E_n = 100$ keV capture cross-section was equal to 0.280 barn. Fission cross-section in the interval from 30 keV to 50 keV was evaluated on measurements results from [Ga75], [Wa80]. Above 50 keV the evaluation from [Ka85] was used. The (n,2n) and (n,3n) reactions cross-sections were calculated on modified version of GNASH code [Yo77]. Neutron transmission coefficients were generated by EGIS and ELIESE-3 cods [Ig72] using deformed and spherical optical potential, respectively.

P. Young, L. Weston, W. Poenitz (LANL, USA) are the authors of ENDF/B-6 (MAT 9437) evaluation; 1989 is a year of evaluation

completion. In thermal region resonance parameters of Reich-Moore were obtained from analysis of experimental data on neutron transmission, fission and capture from [GW71], [GW76], [GW84], [Sp87]. There were also used the results of fission cross-section measurements from [We88]. Experimental fission cross-sections were renormalized to 748 barn at thermal point in accordance with proposed in [26] standard value and with up-dated absolute fission cross-section value from [De72]. To a set of base experiments there were also included the measurements of absolute values of total cross-section at thermal point (1025 ± 6 barn) from [Sp87] and data on fission cross-section measurement from [We88] obtained by time-of-flight method on flight base of 80 meters. In spite of some difficulties because of background presence in last data, their account permitted to improve the evaluations of resonance parameters (resonances at high energy range and their fission widths were identified more carefully). The region of resolved resonances is limited by the energy 2 keV from above. In the range from 1 keV to 2 keV the results from [Sp87] and [To88] were also used. Four resonances at negative energies and six fictitious resonances above 2 keV were used to represent the effect of the external region. Furthermore, for the choice of resonance parameters there were used the results of neutron transmission coefficients measurements from [Ha85], performed on 80 meters transit base with a such high resolution that permit to separate about 80% of resonances at the energies up to 1 keV. There were also used data on fission cross-section measurement from [GW84] and [We84] and the data from [Bl73] having the best resolution which were employed for identification of narrow resonances.

Correlation fitting of correlation data has shown that the renormalizing is appreciably needed for the data from [Bl73], thereat we assume the normalizing is not only the multiplying into correction cross-section coefficients but introducing of smooth "background". In evaluation of radiative width no data on capture cross-section measurements from [GW71] and [GW76] were included in base data set. Careful procedure should be based on analysis of neutron transmission experiment. Evaluated α -values below 500 eV agree approximately with the data from [GW76] and above 500 eV they are 1.7% less than the data from the same work. The region of unresolved resonances is limited by the energy 30 keV from above. Average resonance parameters are given at 74 points on energy. They were obtained by statistical FISINGA code to fit the gross structure of experimental data on total cross-section from [De73] at $E_n < 4$ keV and of selected experimental data on fission cross-section normalized to recommendations of Standard Committee for ENDF/B-6 evaluation were used for their obtaining. Above 4 keV, where the resolution under total cross-section measurement is not large, average total cross-sections were calculated by statistical parameters obtained by parameters of resolved resonance region in [De88] and by parameters of optical model from [La86] fitted to experimental data in the high energy range. The value of 9.46 Fermi was used for scattering radius. Evaluated α -value agrees with experimental data. Above 30 keV the (n,2n) and (n,3n) reaction cross-sections were calculated by GNASH code (statistical method of Houser-Feshbach with preequilibrium evaporation). Capture cross-section was obtained by multiplying of α -value from ENDF/B-5 (2) evaluation into evaluated fission

cross-section. At 30 keV a shift with capture cross-section evaluation in resolved resonance region takes place. Fission cross-section evaluation in the interval from 50 keV to 1.25 MeV differed from recommendations of Standard Committee for ENDF/B-6 by a factor of 1.0007. Between 30 keV and 50 keV fission cross-section was evaluated by parameters of resolved resonances. Above 1.25 MeV fission cross-section evaluations are based on new covariance analysis of all available relative and absolute measurements of ^{239}Pu fission cross-section, including the last measurements of [Li88] and [Me88]. For the covariance analysis of experimental results the GLUCS code was used.

In Tables 6.8-6.14 characteristics of interaction of neutrons with ^{239}Pu obtained by ENDL-82, JENDL-3, ENDF/B-6 and BROND-2 evaluations are compared among themselves and with recommended data.

Table 6.8

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{239}Pu , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
σ_{f0}	785.7	746.7	747.1	748.1	$742.5^{+3.0}_{-3.0}$	1973, [5]
					$748.1^{+2.0}_{-2.0}$	1982, [7]
					$748^{+1.9}_{-1.9}$	1987, [26]
$\sigma_{\gamma0}$	278.0	270.0	271.4	269.3	$268.8^{+3.0}_{-3.0}$	1973, [5]
					$269.3^{+2.2}_{-2.2}$	1982, [7]
					$271.4^{+2.1}_{-2.1}$	1987, [26]

Table 6.9

The values of I_f , I_γ for ^{239}Pu , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	305.2	308.8	307.1	305.3	301_{-10}^{+10}	1973, [5]
					301_{-10}^{+10}	1984, [6]
I_γ	205.2	196.4	196.4	187.7	200_{-20}^{+20}	1973, [5]
					220_{-20}^{+20}	1984, [6]

Table 6.10

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{239}Pu , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.781	0.041	0.007	2.252	0.220	0.080
JENDL-3	1.801	0.051	0.006	2.338	0.290	0.012
ENDF/B-6	1.788	0.039	0.009	2.376	0.257	0.012
BROND-2	1.803	0.055	0.008	2.275	0.130	0.105

Table 6.11

The value of ΔI_f for ^{239}Pu , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	113.3	115.2	113.9	116.7
10	10^2	141.9	143.0	143.0	137.8
10^2	10^3	30.48	30.97	30.98	30.80
10^3	10^4	7.750	7.870	7.520	8.320
10^4	10^5	3.550	4.050	3.610	3.370
10^5	10^6	3.870	3.340	3.720	3.930
10^6	10^7	4.330	4.420	4.390	4.390

Table 6.12

The value of ΔI_γ for ^{239}Pu , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	55.18	50.81	50.89	47.82
10	10^2	110.5	109.5	109.4	104.1
10^2	10^3	30.54	28.04	28.32	26.95
10^3	10^4	7.570	6.330	6.210	7.190
10^4	10^5	1.260	1.360	1.230	1.170
10^5	10^6	0.210	0.340	0.400	0.680
10^6	10^7	0.030	0.030	0.010	0.030

Table 6.13

The value of $\langle \sigma_f \rangle$ for ^{239}Pu , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	30.31	32.38	32.10	32.17
10 10^2	50.62	50.82	50.89	51.57
10^2 10^3	10.20	10.44	10.42	10.34
10^3 10^4	2.769	2.757	2.639	2.859
10^4 10^5	1.590	1.599	1.578	1.578
10^5 10^6	1.610	1.582	1.597	1.608
10^6 10^7	1.934	1.975	1.971	1.950

Table 6.14

The value of $\langle \sigma_\gamma \rangle$ for ^{239}Pu , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	21.05	20.23	20.18	17.35
10 10^2	39.81	39.20	39.16	38.74
10^2 10^3	10.76	8.965	9.153	9.300
10^3 10^4	2.408	2.096	2.065	2.438
10^4 10^5	0.366	0.444	0.412	0.477
10^5 10^6	0.102	0.120	0.106	0.138
10^6 10^7	0.007	0.006	0.003	0.007

For the most part the measurements of ^{239}Pu fission cross-section are relative. As a rule, ^{239}Pu and ^{235}U fission cross-section ratio is measured. On this reason the quality of

^{239}Pu fission cross-section evaluation depends on the accuracy of that of ^{235}U when it is used as a standard. For illustration Table 5.15 is listed where σ_f values for ^{239}Pu both calculated by evaluated data and measured, and also the ratio of $\bar{\sigma}_f$ values for ^{239}Pu and ^{235}U are given.

Table 6.15

The values of $\bar{\sigma}_f(^{239}\text{Pu})$ for $\bar{\sigma}_f(^{239}\text{Pu})/\bar{\sigma}_f(^{235}\text{U})$, in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Measured value	Year of public.
$\bar{\sigma}_f(^{239}\text{Pu})$	1.781	1.801	1.788	1.803	$1.804^{+0.045}_{-0.045}$	1975, [23]
					$1.808^{+0.045}_{-0.045}$	1976, [15]
					$1.805^{+0.036*}_{-0.036}$	1981, [17]
					$1.803^{+0.027}_{-0.027}$	1982, [19]
					$1.814^{+0.026}_{-0.026}$	1983, [9]
					$1.798^{+0.033}_{-0.033}$	1983, [27]
					$1.824^{+0.035}_{-0.035}$	1983, [10]
					$1.824^{+0.035}_{-0.035}$	1984, [11]
					$1.844^{+0.024}_{-0.024}$	1985, [20]
$\bar{\sigma}_f(^{239}\text{Pu})$	-----					
$\bar{\sigma}_f(^{235}\text{U})$	1.446	1.458	1.466	1.458	$1.500^{+0.024}_{-0.024}$	1975, [23]
					$1.500^{+0.024}_{-0.024}$	1976, [15]
					$1.500^{+0.020}_{-0.020}$	1982, [19]
					$1.502^{+0.014}_{-0.014}$	1983, [10]
					$1.502^{+0.014}_{-0.014}$	1984, [11]
					$1.500^{+0.012}_{-0.012}$	1985, [20]

*) - recommended value

6.3 ^{240}Pu

Even-even ^{240}Pu nucleus is a long-lived one ($T_{1/2}=6.563\cdot 10^4$ years, α -decay). This nucleus is produced due to the $^{238}\text{U}(n,\gamma)$ $^{239}\text{U}(\beta^-)$ $^{239}\text{Np}(\beta^-)$ $^{239}\text{Pu}(n,\gamma)$ ^{240}Pu reaction chain. The $^{239}\text{Np}(n,\gamma)$ $^{240\text{gm}}\text{Np}(\beta^-)$ ^{240}Pu branching process is also possible. The half-life for the ^{239}Np ground and metastable states is small enough, therefore one can consider, that neutron radiation capture by ^{239}Np nucleus leads to ^{240}Pu immediate production. Because of small half-life of ^{239}Np (2.355 days) the account of neutron radiation capture process by this nucleus in reactor flux should not lead to significant corrections in ^{240}Pu accumulation. This nucleus disintegrates noticeably in fast neutron flux only. Therefore in thermal and resonance neutron flux radiation capture predominates with production of long lived ^{241}Pu nucleus which even disintegrates by neutrons of all energies. In fission and thermonuclear neutron flux fission process is a main transmutation channel of ^{240}Pu .

The evaluations of characteristics of interaction between neutrons and ^{240}Pu are represented in all versions of evaluated neutron data discussed by us.

R. Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7877) evaluation; 1983 is a year of evaluation completion.

G.Antsipov, V.Kon'shin, A.Klepatskii, V.Zenevich, E.Sukhovitskii (IJE, Belorussia) are the authors of BROND-2 (MAT 9431) evaluation; 1985 is a year of evaluation completion. The region of resolved resonances is between the limits 10^{-5} eV - 1 keV. Single-level Breit-Wigner formula is used. Resonance parameters were evaluated on a basis of data from [Ko68], [We68],

[As66], [By66], [Ho72], [Bo66], [Ca68], [Mi68], [Mo72] with renormalizing to more exact values of neutron widths. In the region of unresolved resonances (1-142 keV) average resonance parameters are given. There was used a two-humped fission barrier conception which leads to a fact that fission width distribution law differs from Porter-Thomas distribution. Evaluation method is given in [An76]. At evaluation in thermal energy region (up to the energy of the first resonance) the interference of potential and resonance scattering for the first resonance was accounted. Above 142 keV fission cross-section evaluation was carried out, mainly, on the base of experimental data from [Ne60], [Fu77], [Be78], [Wh67], [Me81], [Bu81], [Ka78]. Fission cross-section was normalized on ^{235}U one from [Po80]. Method of coupled channels with thoroughly adjusted parameters of optical potential and statistical theory of nuclear reactions were used for evaluation of other types of cross-sections. Transmission coefficients for neutrons were calculated by the method of coupled channels. Parameters of non-spherical optical potential were taken from [Kl80]. Fermi-gas model with collective effect account was used for level density calculations.

T.Hurata, A.Zukeran (NAIG, MITACHI, Japan) are the authors of JENDL-3 (MAT 3944) evaluation. 1990 is a year of evaluation completion. The region of resolved resonances is between the limits 10^{-5} eV and 4 keV. Multilevel Breit-Wigner formula is used. Level parameters at negative energy and at the energy 1.057 eV were revised using recommendations from [6]. At the energy range from 20 to 500 eV neutron and radiative widths were evaluated with the help of experimental data from [Mo72], and at the energies

from 500 eV to 4 keV there were used data from [Ku68]. For resonances with unknown radiative width its value was assumed to be equal to 29.5 meV average value. Below 610 eV fission widths for subthreshold fission were determined by data from [We84], and above 610 eV from [Au75]. In the region of unresolved resonances limited by the energy 40 keV from above, energy dependent average resonance parameters were determined so as to reproduce evaluated cross-sections at this energy range. Fission width were fitted to describe the measurements of average fission cross-sections from [We84]. Below 4 keV background cross-section for radiation capture was introduced. For the region of unresolved resonances background cross-sections were included into smooth cross-section file (MF=3). Above 40 keV the (n,2n) and (n,3n) reaction cross-sections were calculated from neutron emission cross-section and branching coefficients for each channel. Neutron emission cross-section was obtained as a difference between compound nucleus production cross-section and the sum of fission and capture cross-sections. Branching coefficients were obtained with the help of formalism from [Se78]. Between 40 and 100 keV measurement results of average (interval by interval) cross-sections from [We84] normalized on results of fission cross-section evaluation above 100 keV were used for fission cross-section evaluation. At this range fission cross-section evaluation was based on the relative and absolute results of fission cross-section measurements of $^{235,238}\text{U}$, $^{239-241}\text{Pu}$ and ^{197}Au capture cross-section given in [Ka86]. Between 40 and 350 keV fission cross-section evaluation was based on experimental data from [Ho72], [We77] and relative experimental data from

[Wi78], [Wi79] using the results on measurement of fission cross-section for ^{197}Au from [Ka86]. At the same time calculations by statistical model with the help of CASTHY code [Ig75] were also carried out. Above 350 keV capture cross-section was calculated by this code with normalizing on evaluated cross-section at 350 keV. The contribution of direct and collective processes at high energies was account adopting the value for ^{238}U given by [Ki78].

L. Weston, E. Arthur (ORNL, USA) are the authors of ENDF/B-6 (MAT 9440) evaluation. 1986 is a year of evaluation completion. The region of resolved resonances is limited by the energy 5.7 keV from above. Resonance parameters at 1.056 eV were evaluated by data of [Lo70] and [Li85]. Between 20 and 665 eV neutron and radiation widths were evaluated as averaged over data from [We72], [Ko68], [We74], [Ho72] and unchanged data from ENDF/B-5 version. Above 665 eV neutron width were evaluated by data from [We72], [We74]] and extended to 5.7 keV for ENDF/B-6 version. Fission widths were revised over the full energy region of resolved resonances. Data from [We84], [Mi68], [Au75] were used. The region of unresolved resonances is limited by the energy 40 keV from above. Average resonance parameters were taken from [Fr82], [Pe73] and were fitted to description of experimental data from [Mo72], [We77], [Wi78]. For capture cross-section from 2680 eV to 5760 eV there was assumed the background cross-section for effective account of missed resonances. From 40 keV to 300 keV capture cross-section evaluation is based on data from [Mo72], [We77], [Wi78]. Above 300 keV the ENDF/B-4 evaluation was used. Background cross-section is missing for fission one. From 40 to 100 keV fission cross-section evaluation was carried out by data from

[We84], [Bu81]; from 100 keV to 1 MeV it was made according to measurements results of ratio of ^{240}Pu and ^{235}U fission cross-sections from [Bh76] and ENDF/B-5 evaluation for ^{235}U fission cross-section. The evaluation above 1 MeV up to 20 MeV is a compromise between data from [Bh76], [We83], [Ka79], [Gi66], [Sa70], [Me81].

In Tables 6.16-6.22 some characteristics of interaction of neutrons with ^{240}Pu nucleus obtained by ENDL-82, JENDL-3, ENDF/B-6 and BROND-2 evaluations are compared among themselves and with recommended data.

Table 6.16

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{240}Pu , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
σ_{f0}	0.060	0.059	0.064	0.059	$0.03_{-}^{+}0.045$	1973, [5]
					$0.056_{-}^{+}0.030$	1984, [6]
$\sigma_{\gamma0}$	301.5	289.4	287.6	287.8	$289.5_{-}^{+}1.4$	1973, [5]
					$289.5_{-}^{+}1.4$	1984, [6]

Table 6.17

The values of I_f , I_{γ} for ^{240}Pu , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	8.422	7.513	7.586	8.550	8.8	1984, [6]
I_{γ}	8330	8530	8516	8435	$8013_{-}^{+}960$	1973, [5]
					$8100_{-}^{+}200$	1984, [6]

Table 6.18

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{240}Pu , in barn.

Evaluation	f	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.414	0.089	0.005	2.345	0.100	0.150
JENDL-3	1.363	0.090	0.005	2.254	0.373	0.113
ENDF/B-6	1.362	0.076	0.004	2.194	0.460	0.028
BROND-2	1.346	0.086	0.003	2.237	0.367	0.124
measured value	$1.377^{+0.032}_{-0.037}$, [10, 11]					
	$1.310^{+0.037}_{-0.037}$, [12]					

Table 6.19

The value of ΔI_f for ^{240}Pu , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	1.567	1.552	1.644	1.469
10 10^2	0.853	0.276	0.260	0.813
10^2 10^3	0.393	0.301	0.283	0.351
10^3 10^4	0.357	0.314	0.319	0.745
10^4 10^5	0.190	0.217	0.214	0.353
10^5 10^6	0.915	0.863	0.892	0.858
10^6 10^7	4.147	3.990	3.973	3.962

Table 6.20

The value of ΔI_γ for ^{240}Pu , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	8195	8371	8369	8289
10	10^2	103.7	118.9	112.5	109.0
10^2	10^3	25.10	32.60	28.50	28.10
10^3	10^4	4.100	5.300	4.100	6.300
10^4	10^5	1.700	2.700	1.800	1.900
10^5	10^6	0.200	0.400	0.300	0.600
10^6	10^7	0.150	0.100	0.100	0.100

Table 6.21

The value of $\langle \sigma_f \rangle$ for ^{240}Pu , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-32	ENDF/B-6	BROND-2
0.5	10	0.174	0.172	0.182	0.170
10	10^2	0.435	0.097	0.093	0.294
10^2	10^3	0.208	0.218	0.206	0.208
10^3	10^4	0.141	0.105	0.110	0.270
10^4	10^5	0.075	0.090	0.090	0.135
10^5	10^6	0.673	0.638	0.658	0.627
10^6	10^7	1.922	1.840	1.835	1.836

Table 6.22

The value of $\langle \sigma_{\gamma} \rangle$ for ^{240}Pu , in barn.

$E_{\min}$	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
(eV)					
0.5	10	907.1	925.9	925.0	915.3
10	10^2	51.24	59.84	56.83	55.06
10^2	10^3	7.099	9.072	7.651	7.670
10^3	10^4	1.397	1.823	1.518	2.130
10^4	10^5	0.422	0.644	0.579	0.634
10^5	10^6	0.141	0.170	0.143	0.183
10^6	10^7	0.034	0.019	0.018	0.018

6.4 ^{241}Pu

Even-odd ^{241}Pu nucleus is a relatively short-lived one ($T_{1/2}=14.35$ years) transforming into ^{241}Am by β^- -decay. In nuclear reactors ^{241}Pu is produced due to the reaction chains leading first to ^{240}Pu appearance: $^{238}\text{U}(n,\gamma) \rightarrow ^{239}\text{U}(\beta^-) \rightarrow ^{239}\text{Np}(\beta^-) \rightarrow ^{239}\text{Pu}(n,\gamma) \rightarrow ^{240}\text{Pu}$; $^{238}\text{U}(n,2n) \rightarrow ^{237}\text{U}(\beta^-) \rightarrow ^{237}\text{Np}(n,\gamma) \rightarrow ^{238}\text{Np}(\beta^-) \rightarrow ^{238}\text{Pu}(n,\gamma) \rightarrow ^{239}\text{Pu}(n,\gamma) \rightarrow ^{240}\text{Pu}$, or $^{238}\text{Np}(n,\gamma) \rightarrow ^{239}\text{Np}(\beta^-) \rightarrow ^{239}\text{Pu}(n,\gamma) \rightarrow ^{240}\text{Pu}$, or $^{239}\text{Np}(n,\gamma) \rightarrow ^{240}\text{Np}(\beta^-) \rightarrow ^{240}\text{Pu}$; then at neutron capture by ^{240}Pu the ^{241}Pu is produced. ^{241}Pu nucleus disintegrates by neutrons of all energies. In thermal and resonance neutron flux fission reaction with 30% competition from the side of radiation capture is a main transmutation channel; and in fission and thermonuclear neutron flux fission reaction becomes dominant.

The evaluations of characteristics of interaction between neutrons and ^{241}Pu nucleus are given in all versions of evaluated

neutron data compared by us.

R. Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7878) evaluation; 1983 is a year of evaluation completion.

BROND-2 (MAT 9441) evaluation was carried out by G.Antsipov, V.Kon'shin, E.Sukhovitskii, Yu.Porodzinskii, A.Bakhanovich, A.Klepatskii (IJE, Belorussia). 1985 is a year of evaluation completion. At the energy region of resolved resonances (0.25 eV-148.9 eV) resonance parameters were obtained by parameterization (according to Adler-Adler formula) of data on total cross-section from [St83] and by fission cross-section from [Ko71]. At the region of unresolved resonances, limited by the energy 100 keV from above, average resonance parameters were obtained with the help of analysis results from [An82], average parameters from region of resolved resonances, experimental data from fission cross-section in [Fu76], [Be75], [Ka70], [Wh65] and α -value from [We72]. Fission cross-section evaluation up to 0.1 keV was based on data from [Wa64], [Wa66], [Se58], [Ja64], [Ja65], [Ad55], [Wh67], [De59] and above 0.1 keV up to 15 MeV it was based on data from [Fu76], [Be75], [Wh65], [Ka73]. Fission cross-section evaluation was normalized on recommended values of ^{235}U fission cross-section from [Po80]. The (n,2n) and (n,3n) reaction cross-sections were calculated by multiscascade statistical model with allowance for the competition of fission and preequilibrium emission of the first neutron. Used model of nucleus level density accounted collective effects (rotational and vibrational). Capture cross-section was obtained with the help of estimations on optical and statistical models with allowance for the (n,f) process.

L. Weston, R. Wright, H. Derrien (ORNL, USA) are the authors

of ENDF/B-6 (MAT 9443) evaluation completion. The region of resolved resonances is between the limits 10^{-5} and 300 eV. Experimental data used in evaluations were renormalized in the range from 0.02 eV to 0.03 eV in accordance with recommendations of Standard Committee for ENDF/B-6. There were used experimental data: total cross-section from [Yo60], [Si61]; fission cross-section from [Wa76], [We78]; capture cross-section from [We78]. Data on total cross-section from [Yo60], [Si61] and on fission cross-section from [Wa76] were used under correlative R-matrix fitting at the energy range from 0.001 eV to 3 eV. Renormalized data from [Se58] were also included [Se58] for control in the energy range from 0.002 eV to 0.005 eV. The differences between data on total cross-section from [Yo60] and [Si61] are the same as those between calculated and experimental data because of the error of experimental values. There are some discrepancies in data from [We78], which have been already discussed in [We80] during ENDF/B-5 evaluation carrying out. The curve form of fission cross-section from [We78] in the energy range below 0.03 eV differs from data of [Wa76]. Normalized at the range 0.02 eV-0.03 eV data on fission cross-section from these works lead to $\approx 20\%$ mismatch over resonance with the energy 0.26 eV. The same may be saying relative to data on capture cross-section from [We78], which should be renormilized on data from [Si61] or [Yo60] in the energy range over resonance at 0.26 eV. In the region of resolved resonances data on neutron transmission from [Ha72] in the energy range from 0.3 eV to 300 eV together with the data on fission cross-section from [Bl76], [Hi70], [We78] were analyzed. Formalism of Reich-Moore was used. Resonance parameters at negative energies

and with the energies larger 300 eV were chosen so as to reproduce evaluated cross-sections in thermal range and in the range up to 300keV without introducing of a background. The 9.5 Fermi scattering radius was used in this case. Comparison of data on fission cross-section in the region of resolved resonances, carried out in [We78], has revealed a great disagreement among them. It may be caused by neglect of experimental errors, for an example, in case when the background was not fully accounted. In evaluation of resonance parameters this fact was taken into account. Evaluated experimental parameter, as expected, should reproduce the evaluated cross-sections (the fission cross-section, at least) with the accuracy of 3%. This difference is connected with description by Reich-Moore formula with evaluated resonance parameters from [We78], and it may be reduced after renormalizing on data from [Wa76]. Average values of α -quantity obtained by evaluated resonance parameters are 14% less than experimental values from [We78]. Preliminary evaluation together with the data of these authors n capture also used experimental data on fission and neutron transmission in the energy range from 3 eV to 20 eV. In this energy range renormalizing of capture cross-section on 15% was obtained. Absorption experimental data were renormalized by L.Weston on absorption cross-section obtained from total cross-sections evaluated by [Ko68]. Evaluations using data from [Ha72] on neutron transmission lead to the values of radiative width, on the average, 8% less than those obtained from data on neutron transmission from [Ko68]. Data renormalizing on capture cross-section from [We78] to the values following from a new set of neutron width lead to more realistic values of capture

cross-section. In the region of unresolved resonances ENDF/B-5 evaluation was preserved, except capture cross-section reduced by 14%. The region of unresolved resonances is limited by the energy 40.2 keV from above. Above 40.2 keV ENDF/B-5 evaluation reduced by 14% is used for capture cross-section. In the same energy range ENDF/B-5 evaluation based on the results of relative measurements from [Be75], [Ka73], [Sz70] and ENDF/B-5 evaluation for ^{235}U fission cross-section is used for fission cross-section.

Y. Kikuchi, N. Sekine, T. Nakagawa (JAERI, Japan) are the authors of JENDL-3 (MAT 3945) evaluation. 1990 is a year of evaluation completion. The region of resolved resonances is between the limits 1 eV and 100 eV. JENDL-2 evaluation for resonance parameters, which is the modification of JENDL-1 for better agreement with experimental data, is preserved. An addition, in contrast with JENDL-2 the resonance at negative energy and background cross-section for fission and capture processes were included in JENDL-3 evaluation. The region of unresolved resonances is limited by the energy 30 keV from above. Average resonance parameters were adjusted to describe the evaluated fission and capture cross-sections. S_0 , S_1 and fission width are the energy dependent values; scattering radius of 9.8 Fermi, fission width of 0.04 eV and level spacing of 0.85 eV were fixed. From 10^{-5} eV to 1 eV the cross-sections were given point by point: for the total cross-section there were used data from [Sm70]; elastic scattering cross-section was calculated from evaluated resonance parameters of the resonance at negative energy and from resonance contributions at the range above 1 eV; fission cross-section was calculated from data of [Wa76]; capture

cross-section was evaluated as a difference between total cross-section and the sum of fission and elastic scattering cross-sections. Above 30 keV the (n,2n) and (n,3n) reaction cross-sections were calculated by evaporation model. Capture cross-section from 30 keV to 250 keV was evaluated from measurement results of α -value from [We78] and from evaluated values of fission cross-section. Above 250 keV capture cross-section was calculated by CASTHY code [Ig75]). Strength function for γ -quanta was defined so that at the energy 250 eV capture cross-section was equal to 269 mbarn. For fission cross-section above 70 keV in [Ka58] there was carried out the evaluation together with fission cross-section evaluation of $^{235,238}\text{U}$ and $^{240,241}\text{Pu}$. At the same time the measurements results from [Sz70], [Sz73], [Ca78], [Fu78], [Ka73] were used. Below 45 keV to 30 keV JENDL-2 evaluation was preserved. Between 45 keV and 70 keV there is a shift of the last evaluation with that carried out in the range above 70 keV. In Tables 6.23-6.29 the comparison of some characteristics of interaction of neutrons with ^{241}Pu nucleus obtained by evaluated neutron data of ENDL-82, JENDL-3, ENDF/B-6 and BROND-2 among themselves and available recommended data are given.

Table 6.23

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{241}Pu , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
σ_{f0}	996.5	1015	1012	1011	1009^{+8}_{-}	1973, [5]
					$1011.1^{+6.2}_{-}$	1982, [7]
$\sigma_{\gamma0}$	398.6	367.0	363.0	358.2	368^{+10}_{-}	1973, [5]
					$358.2^{+5.1}_{-}$	1982, [7]

Table 6.24

The values of I_f , I_γ for ^{241}Pu , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	579.6	588.2	596.4	613.5	570^{+15}_{-}	1973, [5]
		(590.0) *)			570^{+15}_{-}	1984, [6]
I_γ	225.8	186.9	178.0	204.6	162^{+8}_{-}	1973, [5]
		(190.0)			162^{+8}_{-}	1984, [6]

*) - $E_{\text{max}} = 20 \text{ MeV}$

Table 6.25

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{241}Pu , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.593	0.113	0.046	2.133	0.127	0.605
JENDL-3	1.646	0.115	0.028	2.298	0.114	0.095
ENDF/B-6	1.595	0.081	0.028	2.199	0.105	0.128
BROND-2	1.624	0.068	0.011	2.220	0.500	0.178
measured	$1.616^{+0.080}_{-0.080}$, [10, 11]					:
	$1.744^{+0.054}_{-0.054}$, [12]					

Table 6.26

The value of ΔI_f for ^{241}Pu , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	312.1	326.8	324.7	342.7
10	10^2	191.2	184.7	193.3	185.0
10^2	10^3	49.29	46.97	49.15	53.23
10^3	10^4	12.60	15.36	15.29	17.86
10^4	10^5	6.800	6.170	6.560	7.290
10^5	10^6	3.880	4.350	3.730	3.700
10^6	10^7	3.740	3.860	3.710	3.780

Table 6.27

The value of ΔI_γ for ^{241}Pu , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	124.2	92.83	89.51	104.1
10 10^2	76.40	74.75	71.20	78.95
10^2 10^3	19.55	13.23	12.04	14.70
10^3 10^4	3.160	4.040	3.600	4.890
10^4 10^5	1.690	1.150	1.030	1.410
10^5 10^6	0.710	0.800	0.550	0.540
10^6 10^7	0.100	0.110	0.100	0.120

Table 6.28

The value of $\langle \sigma_f \rangle$ for ^{241}Pu , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	166.6	180.9	180.4	192.1
10 10^2	59.81	56.84	60.91	55.92
10^2 10^3	14.11	16.67	17.24	18.85
10^3 10^4	5.182	5.491	5.444	6.375
10^4 10^5	2.552	2.531	2.470	2.665
10^5 10^6	1.597	1.627	1.618	1.627
10^6 10^7	1.681	1.720	1.648	1.681

Table 6.29

The value of $\langle \sigma_{\gamma} \rangle$ for ^{241}Pu , in barn.

$E_{\min}$	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
(eV)					
0.5	10	66.27	44.54	44.83	50.86
10	10^2	23.86	19.81	19.28	21.59
10^2	10^3	5.511	4.913	4.258	5.631
10^3	10^4	1.117	1.384	1.242	1.622
10^4	10^5	0.656	0.489	0.426	0.596
10^5	10^6	0.251	0.222	0.136	0.114
10^6	10^7	0.026	0.023	0.023	0.017

6.5 ^{242}Pu

Even-even ^{242}Pu nucleus is a long-lived one ($T_{1/2}=3.79 \cdot 10^5$ years, α -decay). It is produced in reactors due to sequential neutron radiation capture by ^{239}Pu , ^{240}Pu , ^{241}Pu nuclei. ^{242}Pu disintegrates noticeably by fast neutrons only. Therefore radiation capture leading to production of short-lived ^{243}Pu nucleus with its transmutation into long-lived ^{243}Am nucleus is the main transmutation channel in thermal and especially resonance neutron flux. Fission is the main ^{242}Pu transmutation channel in fission and thermonuclear neutron flux. Because of inaccessibility the characteristics of interaction of neutrons with ^{242}Pu have been studied worse than, for an example, those previous with $Z=94$. Nevertheless, the evaluations of these characteristics are presented in all compared by us versions of evaluated neutron data. This interest is because of the fact, that ^{242}Pu is an original material in ^{244}Cm generation going by reaction chain:

$^{242}\text{Pu}(n,\gamma)^{243}\text{Pu}(\beta^-)^{243}\text{Am}(n,\gamma)^{244g,m}\text{Am}(\beta^-)^{244}\text{Cm}$. ^{244}Cm nucleus is used as a source of α -radiation and as a radioisotope energy source. ^{242}Pu irradiation for ^{244}Cm production should be carried out in resonance neutron flux.

R. Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7880) evaluation; 1983 is a year of evaluation completion. B.Kon'shin, G.Antsipov, E.Sukhovitskii, L.Bakhanovich, V.Maslov, Yu.Porodzinskii, A.Klepatskii (IJE, Belorussia) are the authors of BROND-2 (MAT 9451) evaluation. 1985 is a year of evaluation completion. The region of resolved resonances is between the limits 10^{-5} eV and 1 keV. Multilevel Breit-Wigner formula is used for cross-section calculations. In the range up to 2 eV the evaluation of resonance parameters for the first resonance was carried out using data on total cross-sections from [Yo70], [Yo71] and data at 2200 m/c in terms of one level at negative energy. In the range up to 1 keV resonance positions, neutron and radiation widths were obtained from [Po73]. Fission width were obtained from experimental data of [Au71]. Average parameters in the region of unresolved resonances (1-200 keV) were derived with the use of data from resolved resonance region and experimental data on radiation capture cross-sections from [Ho75] and fission ones from [Au73]. Fission cross-section was calculated within the limits of two humped fission model. In the range from 0.2 MeV to 15 MeV the evaluation of fission cross-section was based on experimental data from [Al75], [Fu79], [Be78]. Fission cross-section was normalized on ^{235}U one from ENDF/B-5 evaluation. Radiation capture and the (n,2n), (n,3n) reaction cross-sections were calculated by statistical model in terms of competition of fission process and

under the using of level density model with collective modes.

T. Murata, M. Kawai (NAIG, Japan) are the authors of JENDL-3 (MAT 3946) evaluation. 1990 is a year of evaluation completion. The region of resolved resonances is limited by the energy 1.15 keV from above. JENDL-2 evaluation of resonance parameters was modified on a basis of experimental data on fission cross-section from [We85]. Resonance positions corresponded to data from [Mu73]. Neutron and radiation widths were evaluated according to the data from [Po73], [Au73]; fission widths were evaluated from data of [We85]. Scattering radius of 9.9 Fermi, average radiation width of 0.0242 eV were assumed. Two resonances at negative energies for cross-section reproduction in thermal point, recommended in [6] were introduced. The region of unresolved resonances is limited by the energy 40 keV from above. Average resonance parameters were evaluated for reproduction of evaluated cross-sections. Capture cross-section from 6 keV to 210 keV was evaluated on a base of experimental data from [Ho75], [Wi78], [Wi79]; above 210 keV capture cross-section was calculated by statistical model with the help of CASTHY code [Ig75] and was normalized on a shift with the values evaluated at 210 keV. At high energies direct and collective processes with orienting on ^{238}U data published in [Ki78] were accounted. The (n,2n) and (n,3n) reaction cross-sections were calculated as a product of neutron emission cross-section into relative probability of each reaction. Neutron emission cross-section was obtained as the difference between total reaction cross-section and the sum of fission and capture cross-sections. Total reaction cross-section was calculated by spherical optical model. Relative probabilities were calculated by

the model suggested in [Se78]. Below 100 keV energy dependence of fission cross-section was evaluated from data of [Au71] with the following renormalizing on fission cross-section evaluations in the range above 100 keV. The results of relative measurements from [Be78] and ^{235}U fission cross-section evaluation from [Ma87] were used in this region.

Mann, Schenter (HEDL, USA), Madland, Young [LANL, USA], Benjamin [SRL, USA], Howerton [LNLL, USA] are the authors of ENDL/B-6 (MAT 9446) evaluation. 1978 is a year of evaluation completion. The 67 resonances and one bound level describe the cross-sections from 0 eV to 986 eV. Except 2.68 eV level resonance parameters correspond to data from [Mu76]. Level parameters at 2.68 eV were modified so as to preserve the cross-section values and their shape at thermal region in accordance with data from [Yo70b], [Yo71]. In evaluation of capture cross-section data they have been oriented to experimental data from [Bu57], [Ha64], [Be75]. The region of unresolved resonances is limited by the energy 10 keV from above. Average resonance parameters for $l=0$ resonances were obtained from data for resolved resonances. Capture cross-section evaluation above 10 keV were based on calculations by statistical model of compound nucleus. Strength function of γ -quanta was chosen so as to reproduce experimental data from [Ho75]. Contribution of semi-direct processes above 4 MeV was added from preequilibrium cascade calculations using the probability of γ -quanta emission calculated at each stage in [Ar78]. Fission cross-section evaluation below 100 keV was based on experimental data from [Au71]. Above 100 keV the evaluations were based on experimental data from [Be78]. Evaluation of the

(n,2n) and (n,3n) reaction cross-section was carried out numerically by statistical model of compound nucleus.

In Tables 6.30-6.36 some characteristics of interaction of neutrons with ^{242}Pu nucleus obtained from ENDL-82, JENDL-3, ENDF/B-6 and BROND-2, that are compared among themselves and with available recommended data are given.

Table 6.30

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{242}Pu , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
σ_{f0}	0.0009	0.0026	0.0010	0.0010	<0.2	1973, [5]
					<0.2	1984, [6]
$\sigma_{\gamma0}$	18.79	18.79	19.20	18.64	$18.5^{+0.4}_{-0.4}$	1973, [5]
					$18.5^{+0.5}_{-0.5}$	1984, [6]

Table 6.31

The values of I_f , I_γ for ^{242}Pu , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	38.60	4.238	4.234	23.63	5	1973, [5]
		(5.580) ^{*)}			5^{+2}_{-2}	1984, [6]
I_γ	1302	1130	1275	1170	1130^{+60}_{-60}	1973, [5]
		(1130) ^{*)}			1115^{+40}_{-40}	1984, [6]

*) - $E_{\text{max}} = 20 \text{ MeV}$

Table 6.32

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{242}Pu , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.127	0.081	0.011	1.933	0.300	0.350
JENDL-3	1.131	0.085	0.010	2.027	0.433	0.289
ENDF/B-6	1.130	0.070	0.010	1.977	0.417	0.302
BROND-2	1.196	0.071	0.007	2.130	0.256	0.280
measured	$1.082^{+0.017}_{-0.017}$, [9]					

Table 6.33

The value of ΔI_f for ^{242}Pu , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	34.34	0.087	0.0004	0.00004
10	10^2	0.180	0.011	0.057	0.052
10^2	10^3	0.054	0.099	0.109	0.075
10^3	10^4	0.012	0.030	0.055	12.29
10^4	10^5	0.019	0.027	0.028	5.686
10^5	10^6	0.604	0.591	0.601	2.088
10^6	10^7	3.398	3.393	3.384	3.438

Table 6.34

The value of ΔI_{γ} for ^{242}Pu , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	1242	1077	1223	1122
10 10^2	35.40	32.00	30.20	30.20
10^2 10^3	19.60	15.60	15.30	15.40
10^3 10^4	3.500	4.500	4.600	1.300
10^4 10^5	0.370	0.600	0.546	0.145
10^5 10^6	1.370	1.400	1.321	1.311
10^6 10^7	0.100	0.100	0.100	0.100

Table 6.35

The value of $\langle \sigma_f \rangle$ for ^{242}Pu , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	9.628	0.024	0.0001	0.00003
10 10^2	0.091	0.007	0.036	0.032
10^2 10^3	0.015	0.054	0.053	0.042
10^3 10^4	0.006	0.010	0.029	4.326
10^4 10^5	0.012	0.013	0.013	2.345
10^5 10^6	0.475	0.472	0.480	0.681
10^6 10^7	1.574	1.559	1.557	1.609

Table 6.36

The value of $\langle \sigma_{\gamma} \rangle$ for ^{242}Pu , in barn.

$E_{\min}$	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
(eV)					
0.5	10	347.9	300.9	341.8	312.4
10	10^2	19.96	18.21	17.07	17.01
10^2	10^3	6.071	5.113	5.072	5.033
10^3	10^4	1.131	1.637	1.813	0.408
10^4	10^5	0.338	0.535	0.487	0.129
10^5	10^6	0.139	0.142	0.134	0.133
10^6	10^7	0.024	0.020	0.013	0.017

6.6 ^{243}Pu

Even -odd ^{243}Pu nucleus is a short-lived one ($T_{1/2}=4.956$ hours) transforming into ^{243}Am due to β^- decay. ^{243}Pu is produced due to the chain of sequential neutron radiation captures by previous, plutonium isotopes, or by the way of neutron radiation capture by ^{242}Pu nucleus produced as the result of electron capture by ^{242}Am produced in turn at neutron radiation capture by ^{241}Am . In reactor's applications just the ^{243}Pu nucleus ends the transmutation chain of plutonium isotopes, in assumption, that, because of small $T_{1/2}$ capture of neutron by ^{242}Pu leads immediately to ^{243}Am production. Nevertheless, since the ^{244}Pu nuclide following to ^{243}Pu is practically stable, process of neutron radiation capture by ^{243}Pu is of some interest. However, according to existing evaluations in thermal and resonance neutron spectra the fission process of this nucleus is more probable. In fission and thermonuclear neutron flux fission becomes dominant

with possible competition ($\approx 46\%$) in last case, from the side of the (n,2n) reaction leading to ^{241}Pu production. Experimental information about characteristics of interaction between neutrons and ^{243}Pu is rougher poor, and the evaluations of these characteristics are mainly of calculated nature. They are represented only in two compared by us versions of evaluated neutron data, in ENDL-82 and in ENDF/B-6.

R. Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7881) evaluation. 1983 is a year of evaluation completion.

ENDF/B-6 (MAT 9449) evaluation differs from ENDF/B-5 one only by format of data representation.

ENDF/B-5 (MAT 8443) evaluation was carried out by Kinsey, Assembler (BNL, SRL, USA) and was completed in 1976. This evaluation is for the most part the combination of two evaluations, [Be75] and [Ho75]. The region of resolved resonances is between the limits 0 and 101.7 eV. Resonance parameters were evaluated by GENRBAR code [Mc71] and the evaluation was based on conservation of resonance integrals and thermal cross-section evaluations from [Be75]. The region of unresolved resonances is limited by the energy 10 keV from above. Average resonance parameters were evaluated by the same code as those in the region of resolved resonances. Background for fission, capture and elastic scattering cross-sections are equal to zero. Above 10 keV the evaluation of [Ho75] was used for cross-sections.

In Tables 6.37-6.41 the comparison of evaluations of some characteristics of interaction between neutrons and ^{243}Pu nucleus among themselves and with available recommended data are given.

Table 6.37

The values of σ_{f0} , $\sigma_{\gamma 0}$ for ^{243}Pu , in barn.

Value	ENDL-82	ENDF/B-6	Recommend. value	Year of public.
σ_{f0}	180.4	181.4	196^{+16}_{-16}	1973, [5]
			196^{+16}_{-16}	1984, [6]
$\sigma_{\gamma 0}$	268.1	88.11	60^{+30}_{-30}	1973, [5]
			87^{+10}_{-10}	1984, [6]

Table 6.38

The values of I_f , I_γ for ^{243}Pu , in barn.

Value	ENDL-82	ENDF/B-6	Recommend. value	Year of public.
I_f	511.6	558.9	550^{+80}_{-80}	1984, [6]
I_γ	278.1	276.1	270^{+35}_{-35}	1984, [6]

Table 6.39

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{243}Pu , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.077	0.041	0.068	1.900	0.225	0.875
ENDF/B-6	1.077	0.041	0.068	1.900	0.225	0.875

Table 6.40

The values of ΔI_f , ΔI_γ for ^{243}Pu , in barn.

$E_{\min}$ $E_{\max}$ (eV)		ΔI_f		ΔI_γ	
		ENDL-82	ENDF/B-6	ENDL-82	ENDF/B-6
0.5	10	433.2	426.2	222.2	208.0
10	10^2	58.75	97.04	39.90	47.38
10^2	10^3	10.09	23.87	10.09	14.96
10^3	10^4	3.700	5.810	4.390	4.210
10^4	10^5	1.280	1.290	1.020	1.020
10^5	10^6	1.490	1.480	0.460	0.460
10^6	10^7	3.080	3.090	0.020	0.020

Table 6.41

The values of $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{243}Pu , in barn.

$E_{\min}$ $E_{\max}$ (eV)		$\langle \sigma_f \rangle$		$\langle \sigma_\gamma \rangle$	
		ENDL-82	ENDF/B-6	ENDL-82	ENDF/B-6
0.5	10	125.7	125.4	62.01	61.21
10	10^2	15.19	34.63	12.26	16.90
10^2	10^3	3.626	8.086	3.626	5.047
10^3	10^4	1.359	1.898	1.708	1.583
10^4	10^5	0.636	0.637	0.370	0.370
10^5	10^6	0.523	0.523	0.102	0.102
10^6	10^7	1.396	1.396	0.007	0.007

6.7 ^{244}Pu

Even-even ^{244}Pu nucleus is a practically stable one ($T_{1/2}=8.08\cdot 10^7$ years, α -decay). It can be produced in reactors due to neutron radiation capture by ^{243}Pu nuclei. However this process is hardly probable at reactor's intensities of neutron flux density because of small $T_{1/2}$ (equal to 5 hours) for last nucleus. In comparison with the other plutonium isotopes ^{244}Pu nucleus has a small capture cross-section and zeroth (from evaluations) fission cross-section in thermal neutron flux, that is, in thermal reactors it is the inert material. Some capture increase and the appearance of undeshreshold fission may take place in resonance neutron flux. Because of ^{244}Pu inaccessibility the characteristics of interaction of neutrons with this nucleus were evaluated by calculation mainly, and among the versions of evaluated neutron data considered by us they are presented only in ENDF/B-6. ENDF/B-6 (MAT 9452) evaluation differs from ENDF/B-5 (MAT 8444) one only by format of data representation. The last evaluation was completed in 1978. It was carried out by F. Mann, R. Schenter, Benjamin, McCrosson (HEDL, SRL, USA). The region of resolved resonances is limited by the energy 249.2 eV from above. Resonance parameters were evaluated by GENRBAR code [Mc71] with the use of data from SRL. The region of unresolved resonances is limited by the energy 10 keV from above. Average resonance parameters were evaluated on the same basis set as those of resolved resonances were. Above 10 keV capture cross-section was calculated by statistical model from [Ma76] and [Ma77]. Fission cross-section with the use of data from [Be75] was calculated by the same model.

In Tables 6.42-6.44 some characteristics of interaction between neutrons and ^{244}Pu nucleus are given.

Table 6.42

The values of σ_{f0} , $\sigma_{\gamma 0}$, I_f , I_γ for ^{244}Pu , in barn.

Evaluation	σ_{f0}	$\sigma_{\gamma 0}$	I_f	I_γ
ENDF/B-6	0	1.833	3.554	105.6
Recommended				
value, year of	-	$187^{+0.1}_{-0.1}$, 1973, [5]	-	43^{+4}_{-4} , 1973, [5]
publication				$40.6^{+2.9}_{-2.9}$, 1984, [6]

Table 6.43

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{244}Pu , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDF/B-6	1.006	0.017	0.030	1.695	0.269	0.380

Table 6.44

The values of ΔI_f , ΔI_γ , $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{244}Pu , in barn, for ENDF/B-6.

$E_{\min}$ (eV)	$E_{\max}$	ΔI_f	ΔI_γ	$\langle \sigma_f \rangle$	$\langle \sigma_\gamma \rangle$
0.5	10	0	0.482	0	0.099
10	10^2	0	78.60	0	33.17
10^2	10^3	0.002	22.90	0.001	6.376
10^3	10^4	0.025	2.990	0.015	0.901
10^4	10^5	0.039	0.501	0.023	0.151
10^5	10^6	0.452	0.118	0.237	0.036
10^6	10^7	3.036	0.010	1.381	0.003

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^{238}Pu

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7. Isotopes of $^{241-244}\text{Am}$, $^{242\text{m}}, ^{244\text{m}}\text{Am}$.

7.1 ^{241}Am .

Odd-even ^{241}Am nucleus is relatively a long-lived one ($T_{1/2}=432.7$ years) transforming into practically stable ^{237}Np due to α - decay. ^{241}Am is produced by sequential neutron capture by plutonium isotopes up to ^{241}Pu and by β - decay of the last nucleus with $T_{1/2}=14.35$ years. Because of ^{242}Am decay by orbital electron capture (the branch is about 17%) with $T_{1/2} = 16.02$ hours into ^{242}Pu , following close transmutation chain may take place - $^{241}\text{Am} (n, \gamma) ^{242\text{g}}\text{Am} (3.3) ^{242}\text{Pu} (n, 2n) ^{241}\text{Pu} (\beta) ^{241}\text{Am}$. $^{242\text{g}}\text{Am}$ is produced under capture of a neutron by ^{241}Am . β - decay (the branch is about 84%) into ^{242}Cm nucleus, used in α - radiation sources and radioisotope energy ones, is another channel of $^{242\text{g}}\text{Am}$ decay. Therefore ^{241}Am is the original material for ^{242}Cm production. In thermal and resonance neutron flux ^{241}Am should be burnt out mainly at neutron capture, thereat both the $^{242\text{g}}\text{Am}$ and with, approximately, lesser by a factor of 10 probability $^{242\text{m}}\text{Am}$ are produced. $^{242\text{m}}\text{Am}$ nucleus decays by γ - quanta emission (with $T_{1/2}=141$ years) into $^{242\text{g}}\text{Am}$ and it has the record fission cross-section both in the region of thermal and resonance neutrons. Fission process could be the main ^{241}Am transmutation process in the region of fission and thermonuclear neutrons.

No data in literature on differential measurements of neutron capture cross-section of ^{241}Am at thermal point are present, there are only the results of integral measurements: the measurements either on thermal reactor columns by activation method, or on reactor spectra with the neutron average energy of 0.0253 eV. For production

cross-sections of ^{242g}Am and ^{242m}Am nuclei there is an agreement of various authors data within the limits 7-15%, but they are, in average, 30% above than those proposed in [6]. Summary production cross-sections of ^{242}Am isomers are in good agreement with measured independently neutron total cross-section capture by ^{241}Am nucleus. It permits to suppose that capture cross-sections proposed in [6] are greatly undervalued. The results of I_f measurements for ^{241}Am by integral methods or by activation one with Cb shield for more or less 'clean' samples of ^{241}Am , or by the method of isotope accumulation in reactor fuel (measurements of the 2-nd and 3-d groups) within the limits of $\pm 10\%$ agree to one another, but they are below than those recommended in [6]. ^{241}Am fission cross-section was measured at thermal point both by differential and integral methods. The results of various authors are in good agreement with themselves and with recommended values from [6]. As a rule there is good agreement of evaluated by various methods I_f value including a differential one; but the value proposed in [6] is seemed to be conservative, apparently, because of conservative upper integration limit. Evaluations of characteristics of interaction between neutrons and ^{241}Am nucleus are given in all versions of evaluated neutron data compared by us.

R.Howerton is the author of ENDL-82 (MAT 7882) evaluation. 1983 is a year of evaluation completion. A.Blokhin, V.Maslov (FEI Russia, IJE, Belorussia) are the authors of BROND-2 (MAT 9541) evaluation. 1990 is a year of evaluation completion. The region of resolved resonances is limited by the energy 150 eV from above. For cross-section recovering from resonance parameters multilevel

Breit-Wigner formula is used. Data from [Ab81] and a resonance at negative energy were used for resonance parameter evaluation. The region of unresolved resonances is limited by the energy 30 keV from above; there were used the evaluations from JENDL-3. Average parameters were determined so as to reproduce measurement results of capture cross-section from [Va86] and fission cross-section from [Da83]. $R = 9.37$ Fermi, $\bar{\Gamma}_{\gamma} = 0.044$ eV, $D = 0.4$ eV were energy independent. S_0 ($1.088 \cdot 10^{-4}$ at 150 eV, $0.79 \cdot 10^{-4}$ at 30 keV), Γ_f (0.24 MeV at 150 eV, 0.30 MeV at 30 keV) were energy dependent parameters. Above 30 keV JENDL-3 evaluation was used for fission cross-section. Capture cross-section was calculated by statistical model with the help of CASTHY code, thereat the strength function of γ - quanta was chosen so that at 60 keV capture cross-section is to be equal to 1.7 barn. At the same energy range fission cross-section was evaluated from data of [Da83] and [Ka79]. The (n,2n) and (n,3n) reaction cross-sections were calculated by statistical model from [Ig88].

T.Nakagawa (JAERI, Japan) is the author of JENDL-3 (MAT 3951) evaluation. 1988 is a year of evaluation completion. The region of resolved resonances is limited by the energy 150 eV from above. Multilevel Breit-Wigner formula was used. Data from [De75] were used in evaluation of resonance parameters. There were adopted five resonances at negative energy. The region of unresolved resonances is limited by the energy 30 keV from above. The rest characteristics of this range are given in a previous paragraph where BROND-2 is represented. Above 30 keV capture cross-section was calculated by statistical model with the help of CASTHY code [Ig75] and strength function of γ -quanta was chosen so that at 60 keV capture

cross-section was equal to 1.7 barn. At the same energy range fission cross-section was evaluated from data of [Da83]. The (n,2n) and (n,3n) reaction cross-sections correspond to JENDL-2 evaluation were obtained by calculations with the help of evaporation model.

Zhou Delin, Gu Fuhua (CNDC,USA) are the authors of ENDF/B-6 (MAT 9543) evaluation. 1988 is a year of evaluation completion. The region of resolved resonances is limited by the energy 150 eV from above Breit-Wigner formula of isolated levels is used. Parameters of resolved resonances were determined by the following way. It was noted, that the parameters of first few well studied resonances (for example, at 0.308, 0.576, 1.276, 1.928, 2.372, 2.372, 2.358 eV energies) from [De75] (transmission method) [Ka76] (transmission method) and [We76] (absorption at normalizing on capture cross-sections, 582 barn, in thermal point) are well agree among themselves. This fact provided the basis for evaluation of resonance parameters as average weighted ones without account of 8% systematic error is absorption measurements from [We75]. The region of unresolved resonances extends to 30 keV. To obtained average resonance parameters there were used data from [Ki82] corrected to capture cross-section measurements from [We76], [Va85], [Wi80], [De75] and to fission cross-section evaluations from [Da83] averaged over a number of energy intervals. Above 30 keV capture cross-section was calculated by statistical model and the (n,2n), (n,3n) reaction cross-sections were calculated by evaporation one. At $E_n > 1$ keV fission cross-section was evaluated in [Gu85] and revised in [Th88] in terms of new experimental data from [Da83].

In Tables 7.1-7.7 the comparison of some characteristics of

interaction of neutrons with ^{241}Am obtained from ENDL-82, JEDL-3, ENDF/B-6 and BROND-2 among themselves and with recommended data are given.

Table 7.1

The values of $\sigma_{\gamma 0}$, σ_{f0} for ^{241}Am , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
$\sigma_{\gamma 0}$	525.5	600.1	620.6	589.2	$832^{+}_{-}20$	1973, [5]
					$587^{+}_{-}12$	1984, [6]
					$824^{+}_{-}20^{*)}$	1993, [3]
					$820^{+}_{-}32^{**)}$	1964, [28]
σ_{f0}	3.282	3.018	3.148	3.108	$3.15^{+}_{-}0.10$	1973, [5]
					$3.20^{+}_{-}0.09$	1984, [6]
					$3.15^{+}_{-}0.13$	1993, [3]
					$3.0^{+}_{-}0.3$	1993, [3]

*) - thermal column measurements;

**) - measurement (or measurements in case of σ_{f0}) on reactor spectrum with the average neutron energy of 0.0253 eV.

Table 7.2

The values of I_f , I_γ for ^{241}Am , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	14.75	12.16	13.18	12.25	21_{-2}^{+2}	1973, [5]
					$14.4_{-1.0}^{+1.0}$	1984, [6]
					$14.1_{-0.9}^{+0.9^{*)}}$	1983, [29]
					$21.9_{-1.7}^{+1.7^{**})}$	1993, [3]
					$22.2_{-2.1}^{+2.1^{***})}$	1993, [30]
I_γ	1507	1308	1388	1417	1477	1973, [5]
					1425_{-100}^{+100}	1984, [6]
					$1626_{-143}^{+143^{**})}$	1973, [31]
					$1469_{-147}^{+147^{***})}$	1971, [30]

*) - $E_{\text{max}} = 1 \text{ MeV}$

**) - measurements by activation method

***) - reactor measurements

Table 7.3

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{241}Am , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.502	0.157	0.016	2.686	0.117	0.100
JENDL-3	1.388	0.223	0.001	2.660	0.262	0.047
ENDF/B-6	1.419	0.361	0.0002	2.670	0.204	0.001
BROND-2	1.395	0.223	0.003	2.290	0.340	0.005

Table 7.4

The value of ΔI_f for ^{241}Am , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	7.262	6.326	6.878	6.229
10	10^2	1.504	0.633	0.936	0.769
10^2	10^3	0.548	0.302	0.322	0.302
10^3	10^4	0.143	0.084	0.089	0.084
10^4	10^5	0.074	0.035	0.035	0.036
10^5	10^6	0.432	0.396	0.398	0.393
10^6	10^7	4.791	4.387	4.526	4.435

Table 7.5

The value of ΔI_γ for ^{241}Am , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	1262	1096	1184	1191
10	10^2	171.7	131.1	127.1	145.5
10^2	10^3	53.40	57.80	53.90	57.90
10^3	10^4	13.90	16.00	15.50	16.00
10^4	10^5	4.761	5.423	5.530	5.423
10^5	10^6	1.239	1.577	2.102	1.577
10^6	10^7	0.100	0.100	0.300	0.100

Table 7.6

The value of $\langle \sigma_f \rangle$ for ^{241}Am , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	1.428	1.427	1.465	1.338
10 10^2	0.679	0.222	0.348	0.280
10^2 10^3	0.191	0.106	0.108	0.106
10^3 10^4	0.051	0.030	0.030	0.030
10^4 10^5	0.030	0.015	0.015	0.015
10^5 10^6	0.360	0.330	0.330	0.330
10^6 10^7	2.241	2.025	2.116	2.059

Table 7.7

The value of $\langle \sigma_\gamma \rangle$ for ^{241}Am , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	232.5	187.6	193.1	194.5
10 10^2	62.53	48.32	47.35	52.72
10^2 10^3	18.20	19.87	17.96	19.87
10^3 10^4	4.864	5.446	5.304	5.446
10^4 10^5	1.733	1.922	1.967	1.922
10^5 10^6	0.451	0.559	0.811	0.559
10^6 10^7	0.020	0.021	0.054	0.021

7.2 ^{242}Am

^{242g}Am is a short-lived isomer with $T_{1/2}=16.02$ hours. It decays into two channels transforming into ^{242}Cm due to β -decay and into ^{242}Pu due to orbital electron capture. ^{242g}Am is produced at neutron

capture by ^{241}Am and at γ -decay of $^{242\text{m}}\text{Am}$ isomer, which in its turn is also produced at neutron capture by ^{241}Am . The probability of $^{242\text{g}}\text{Am}$ production in this process is approximately ten times as much as a probability of $^{242\text{m}}\text{Am}$ production. Fission reaction, the probability of which in thermal and resonance neutron flux is about ten times as much as the probability of radiation neutron capture, is the main channel of $^{242\text{g}}\text{Am}$ burning out.

Experimental data about characteristics of interaction of neutrons with $^{242\text{g}}\text{Am}$ are rougher poor. The only integral capture cross-section measurement at thermal point performed on reactor is known. Fission cross-section at thermal point was measured by integral methods only. Various authors data are satisfactorily agree between themselves and with recommended values from [6]. Experimental data and proposed values for I_γ are missing. There is one I_γ experimental value obtained by integral method. Nevertheless, evaluations of characteristics of interaction between neutron and $^{242\text{g}}\text{Am}$ isomer are given in all compared by us versions of evaluated neutron data. In preliminary analysis we have made a conclusion that ENDL-82 ^{242}Am evaluation should be attributed to $^{242\text{m}}\text{Am}$, and therefore the corresponding values will be given in chapter devoted to $^{242\text{m}}\text{Am}$.

JENDL-3 (MAT 3952) evaluation agrees with JENDL-2 one. T.Nakagawa, S.Igarasi (JAERI, Japan) are the authors of the last. 1980 is a year of evaluation completion. Cross-sections in resonance region are given pointwise. Below 0.225 eV $1/v$ dependence is used for fission and capture cross-section. Above 0.225 eV capture cross-section was calculated by statistical model with the help of CASTHY code [Ig75] in terms of the competition from the side of

fission and the (n,2n), (n,3n) reaction cross-sections. In this case the values of $\bar{\Gamma}_{\gamma} = 0.05$ eV and $D = 0.45$ eV were used. Fission cross-section at the same energy range was evaluated by empirical formula for ^{242m}Am with the shift in energy by 49 keV. The (n,2n) and (n,3n) reaction cross-sections were calculated by evaporation model from [Pe65].

The ENDF/B-6 (MAT 9546) evaluation differs from ENDF/B-5 one only by format of data representation.

F.Mc Crosson, R.Benjamin (SRL,USA) are the authors of ENDF/B-5 evaluation. 1975 is a year of evaluation completion. The region of resolved resonances is limited by the energy 101.2 eV from above. Resonance parameters were determined by GENRPAR code [Mc71] from SRL data set for the average level spacing of 1.2 eV. The region of unresolved resonances is limited by the energy 10 keV from above. For obtaining of the average resonance parameters there was used the same data set basis as in the region of resolved resonances. No cross-section data above 10 keV were in the file being at out disposal.

BROND-2 (MAT 9542) evaluation was carried out by A.Blokhin and V.Maslov (FEI, Russia; IJE, Belorussia); it was completed in 1990. In resonance region the cross-sections are given point by point. Below 30 keV the $1/v$ dependence with normalizing by 5100 barn at thermal point was adopted. Above 30 keV capture cross-section was calculated by STAPRF code [Ma88] (statistical model); thereat the strength γ -function was chosen so that capture cross-section at 30 keV was equal to 0.74 barn. Below 30 keV the $1/v$ law with normalizing by 2150 barn at thermal point was also adopted for fission cross-section.

Above 30 keV fission cross-section was calculated by STAPRF code using data base from [Br84]. The (n,2n) and (n,3n) reaction cross-sections were calculated by statistical model from [Ig88].

In Tables 7.8-7.14 some characteristics of interaction of neutrons with ^{242}gAm obtained from JENDL-3, ENDF/B-6 and BROND-2 are compared among themselves and with recommended values.

Table 7.8

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{242}gAm , in barn. :

Value	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
σ_{f0}	2100	2269	2150	2900^{+100}_{-100}	1973, [5]
				2100^{+120}_{-120}	1984, [6]
				$2150^{+200}_{-200}^{*)}$	1993, [3]
				$2195^{+170}_{-170}^{**})$	1993, [3]
$\sigma_{\gamma0}$	5500	252.1	5100	5100^{+1550}_{-1550}	1993, [3]

*) - thermal column measurements

**) - reactor spectra measurements

Table 7.9

The values of I_f , I_γ for ^{242}gAm , in barn.

Value	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	1257	622.2	1163	$< 300^{*)}$	1967, [32]
I_γ	390.8	72.10	1228	-	

*) - measurement by activation method with Cd shield

Table 7.10

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{242}gAm , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
JENDL-3	1.759	0.202	0.006	2.427	0.131	0.139
BROND-2	1.838	0.114	0.003	2.550	0.281	0.014

Table 7.11

The value of ΔI_f for ^{242}gAm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	JENDL-3	ENDF/B-6	BROND-2
0.5 10	969.5	443.6	869.5
10 10^2	190.8	127.5	188.8
10^2 10^3	60.20	38.64	64.70
10^3 10^4	19.50	12.46	22.20
10^4 10^5	8.916	0	8.500
10^5 10^6	4.284	0	4.800
10^6 10^7	4.100	0	4.100

Table 7.12

The value of ΔI_γ for ^{242}gAm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	JENDL-3	ENDF/B-6	BROND-2
0.5 10	312.1	49.29	1043
10 10^2	55.3	14.17	141.7
10^2 10^3	16.26	6.788	33.10
10^3 10^4	4.520	1.851	7.700
10^4 10^5	1.820	0	2.189
10^5 10^6	0.610	0	0.378
10^6 10^7	0.190	0	0.100

Table 7.13

The value of $\langle \sigma_f \rangle$ for ^{242}gAm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	JENDL-3	ENDF/B-6	BROND-2
0.5 10	228.1	121.4	209.9
10 10^2	66.98	44.58	67.35
10^2 10^3	21.17	13.50	23.08
10^3 10^4	6.962	4.324	7.913
10^4 10^5	3.190	0	3.111
10^5 10^6	1.975	0	2.062
10^6 10^7	1.890	0	1.859

Table 7.14

The value of $\langle \sigma_{\gamma} \rangle$ for ^{242g}Am , in barn.

$E_{\min}$ $E_{\max}$ (eV)	JENDL-3	ENDF/B-6	BROND-2
0.5 10	69.15	13.49	223.2
10 10^2	19.26	4.953	47.03
10^2 10^3	5.587	2.359	10.97
10^3 10^4	1.550	0.569	2.561
10^4 10^5	0.561	0	0.675
10^5 10^6	0.410	0	0.254
10^6 10^7	0.033	0	0.015

7.3 ^{242m}Am

In contrast with ^{242g}Am the ^{242m}Am nucleus is a long-lived one with $T_{1/2} = 141$ years. It transforms into the ground state of ^{242g}Am by γ - quanta emission. ^{242m}Am is produced at neutron capture by ^{241}Am nucleus but with a probability lesser, approximately, by a factor of 10, than ^{242g}Am has. Fission reaction, which probability is record in resonance and especially in thermal neutron flux, is the main transmutation channel in neutron flux of all energies. Although, an estimated the cross-section of capture of thermal neutrons by ^{242g}Am is greater by several fold than the same characteristics for ^{242m}Am , neutron capture by ^{242m}Am should contributed mainly to ^{243}Am

production under reactor's conditions, because it is a long-lived isomer in comparison with ^{242g}Am with the half-life of 16 hours.

The only integral measurement of $\sigma_{\gamma 0}$ on isotope accumulation in reactor is known. It agrees with the values proposed in [5] within the limits of errors. Fission process at thermal point has been studied both by differential and integral methods. Various authors data are satisfactorily agree among themselves and with σ_{f0} value proposed in [6]. Experimental data about I_{γ} are missing. The results of some authors devoted to I_f measurements by differential and integral methods are known. There is a disagreement between measurement results of these different methods. However, the I_f evaluation according to measurement results by differential method overlaps with the proposed values from [6] within the limits of errors.

The evaluations of characteristics of neutrons with ^{242m}Am are given in all versions of evaluated data discussed. Thereat, it must be noted, that the ENDL-82 evaluation for ^{242}Am referred by us to ^{242m}Am .

R. Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7883) evaluation. 1983 is a year of evaluation completion.

T. Nakagawa (JAERI, Japan) is the author of JENDL-3 (MAT 3933) evaluation. 1988 is a year of evaluation completion. The region of resolved resonances is limited by the energy 20 eV from above. Parameters of 48 levels were obtained in [Br84]. Single-level Breit-Wigner formula is used. The region of unresolved resonances is limited by the energy 30 keV from above. Average resonance parameters were defined so as to reproduce fission cross-section evaluation from [Br84]. Background cross-section was adopted near 30 keV for fission reaction. There were used the values of $\bar{\Gamma}_{\gamma} = 0.05$ eV, $\bar{\Gamma}_f = 1.28$ eV,

$\bar{D} = 0.4$ eV, $R = 9.59$ Fermi, $S_0 = 1.074 \cdot 10^{-4}$. S_1 is energy dependent value. Above 30 keV capture cross-section was calculated by statistical model with the help of CASTHY code [Ig75]. The strength function of γ -quanta corresponded to the $\bar{\Gamma}_\gamma = 0.05$ eV and $\bar{D} = 0.4$ eV values. The (n,2n) and (n,3n) reaction cross-sections correspond to JENDL-2 evaluation obtained by calculations with the help of evaporation model from [Pe65]. At $E_n > 30$ keV fission cross-section was obtained due to approximation by cube splines of measurement results from [Br84].

F.Mann, R.Benjamin, R.Howerton and et al. (HEDL, SRL, LNLL, USA) are the authors of ENDF/B-6 (MAT 9547) evaluation. The evaluation was completed in 1978. The region of resolved resonances is between the limits 0 and 3.55 eV. Resonance parameters for 6 resolved resonances were taken from [Bo68]. The region of unresolved resonances is limited by the energy 10 keV from above. Average resonance parameters were taken from the same work. Above 10 keV capture cross-section was calculated by statistical model described in [Ma76]. Fission cross-section with orientation to data from [Bo68], [Sc67] and the (n,2n) and (n,3n) reaction cross-sections were calculated by the same model.

BROND-2 (MAT 9543) was carried out by A. Blokhin and V.Maslov (FEI, Russia, IJE, Belorussia) and was completed in 1990. The region of resolved resonances is limited by the energy 20 eV from above. Multilevel Breit-Wigner formula is used. Resonance parameters correspond to evaluations from [Ab81], except the first resonance, the parameters of which were slightly changed to reproduce cross-sections at thermal point. The region of unresolved resonances extends to 30 keV. For average resonance parameters the JENDL-3

evaluation was used. Below 30 keV background cross-section for capture was equal to zero. Above 30 keV capture cross-section was calculated by STAPRF code from [Ma88], thereat the strength function of γ - quanta was chosen so that capture cross-section was equal to 0.74 barn at 30 keV. At the same energy range fission cross-section was evaluated by data from [Br84] according to the data from [Ma88]. The (n,2n) and (n,3n) reaction cross-sections were calculated by statistical model represented in [Ig88].

In Tables 7.15-7.21 some characteristics of interaction of neutrons with ^{242m}Am obtained from various evaluations are compared among themselves and with recommended data.

Table 7.15

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{242m}Am , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
σ_{f0}	6328	6409	6620	6972	6600^{+300}_{-}	1973, [5]
					6950^{+280}_{-}	1984, [6]
					$6713^{+302}_{-}^{*)}$	1993, [3]
					$6748^{+488}_{-}^{**})$	1993, [3]
					$6800^{+1360}_{-}^{***})$	1972, [33]
$\sigma_{\gamma0}$	1994	1254	1342	1619	1400^{+860}_{-}	1973, [5]
					1344	1984, [34]
					$1650^{+330}_{-}^{***})$	1972, [33]

*) - differential measurements

**) - measurements by activation method with Cd shield

***) - reactor measurements

Table 7.16

The values of I_f , I_γ for ^{242m}Am , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	1540	1593	1899	1539	1800^{+65}_{-65}	1984, [6]
					$1677^{+85}_{-85}^{*)}$	1993, [3]
					$2070^{+100}_{-100}^{**)}$	1993, [3]
I_γ	164.0	246.1	289.0	247.0	230^{+100}_{-100}	1981, [35]

*) - differential measurements

**) - integral measurements by activation method with Cd shield

Table 7.17

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{242m}Am , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.825	0.095	0.028	2.430	0.133	0.227
JENDL-3	1.837	0.113	0.006	2.550	0.131	0.139
ENDF/B-6	2.214	0.019	0.017	2.600	0.194	0.350
BROND-2	1.836	0.113	0.003	2.550	0.281	0.014

Table 7.18

The value of ΔI_f for ^{242m}Am , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	1263	1282	1393	1295
10 10^2	186.5	211.0	346.8	162.5
10^2 10^3	54.90	61.60	107.3	48.20
10^3 10^4	19.20	21.00	32.00	16.30
10^4 10^5	7.800	6.811	10.400	6.700
10^5 10^6	4.475	4.548	4.700	4.600
10^6 10^7	4.100	4.100	5.000	4.100

Table 7.19

The value of ΔI_γ for ^{242m}Am , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	136.0	194.4	203.7	195.6
10 10^2	18.02	32.93	60.20	32.88
10^2 10^3	5.530	12.41	18.37	12.40
10^3 10^4	2.760	3.920	5.290	3.920
10^4 10^5	1.670	1.330	1.130	1.330
10^5 10^6	0.330	1.010	0.126	1.010
10^6 10^7	0.110	0.090	0.020	0.090

Table 7.20

The value of $\langle \sigma_f \rangle$ for ^{242m}Am , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	230.6	232.8	345.9	233.7
10 10^2	66.24	77.73	121.0	55.44
10^2 10^3	18.99	20.91	37.27	16.99
10^3 10^4	6.831	7.450	10.97	5.940
10^4 10^5	3.148	3.088	3.684	3.023
10^5 10^6	2.029	2.062	2.380	2.062
10^6 10^7	1.873	1.859	2.264	1.859

Table 7.21

The value of $\langle \sigma_\gamma \rangle$ for ^{242m}Am , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	26.19	35.36	52.53	35.31
10 10^2	6.132	12.95	20.97	12.94
10^2 10^3	2.001	4.335	6.347	4.335
10^3 10^4	1.126	1.401	1.800	1.401
10^4 10^5	0.544	0.433	0.368	0.434
10^5 10^6	0.165	0.320	0.040	0.320
10^6 10^7	0.026	0.015	0.005	0.015

7.4 ^{243}Am

Odd-even ^{243}Am is a long-lived nucleus ($T_{1/2} = 7380$ years, α -decay). ^{243}Am is produced under neutron capture by ^{242g}Am and ^{242m}Am nuclei. In this case, since $T_{1/2}(^{242m}\text{Am}) \gg T_{1/2}(^{242g}\text{Am})$ neutron

capture by ^{242m}Am is the main channel under reactor's conditions. In thermal and resonance neutron flux ^{243}Am nucleus capture neutrons mainly with ^{244g}Am and ^{244m}Am isomer production; this reaction comes most intensively in resonance neutron flux where first the ^{244m}Am generation go on [5,6]. In fission and thermonuclear neutron flux ^{243}Am undergo fission dominantly. Capture cross-section in thermal point for ^{243}Am was measured by integral methods only. The agreement of data obtained from various references is at a level of $\pm 10\%$ [3]. The discrepancies between fission cross-sections at thermal point measured by different authors may reach about 300%. Consequently, if the fission cross-section value at thermal point, proposed in [3], agrees with the most earlier recommendation from [5], this from [3] is less, than that recommended in later work [6], approximately, by a factor of 3. The evaluations of characteristics of interaction of neutrons with ^{243}Am are represented in all versions of evaluated neutron data compared by us.

R. Howerton (LNLL,USA) is the author of ENDL-82 (MAT 7884) evaluation. 1983 is a year of evaluation completion.

The JENDL-3 (MAT 3954) evaluation was carried out by T. Nakagawa (JAERI, Japan). It was completed in 1988. The region of resolved resonances extends to 215 eV. There were used multilevel Breit-Wigner formula and JENDL-2 evaluation of resonance parameters, which was carried out according to the data from [Si74]. The exception is the fission width that was changed according to the data from [Kn86]. The region of unresolved resonances is limited by the energy 30 keV from above. JENDL-2 evaluation is used for average resonance parameters. The values of $S_0 = 0.93 \cdot 10^{-4}$, $S_1 = 2.44 \cdot 10^{-4}$, $R = 9.34$ Fermi were

obtained by optical model estimations, and the values of $\bar{D} = 0.67$ eV, $\bar{\Gamma}_{\gamma} = 0.039$ eV and $\Gamma_f = 0.00012$ eV were obtained by resolved level parameters. Above 30 keV capture cross-section was calculated by CASTHY code [Ig75] (statistical model), strength function of γ -quanta was determined so as to reproduce capture cross-section value of 2.2 barn at 30 keV from [We85]. In the interval between 30 and 100 keV fission cross-section evaluation was performed as a smooth extension of evaluation in the region of unresolved resonances. From 100 keV to 10 MeV spline approximation of data from [Ka87] was carried out, and above 10 MeV data from [Be81] were used. The (n,2n) and (n,3n) reaction cross-section evaluation was preserved the same as in JENDL-2 evaluation.

L.Weston, F.Mann, R. Howerton B. Schenter (ORNL, HEDL, LNL, USA) are the authors of ENDF/B-6 (MAT 9549) evaluation. 1988 is a year of evaluation completion. The region of resolved resonances is limited by the energy 250 eV from above. 220 levels have been accounted. To obtain level parameters there were used the results of total cross-section measurements from [Si74], the evaluations of capture thermal cross-section from [6] and the results of measurements of fission thermal cross-section from [Wa88]. The evaluation of fission widths was based on measurement results from [Kn88]. The region of unresolved resonances extends to 42 keV. The evaluation of average resonance parameters was based on those from [Fr82], [We85]. Above 42 keV and up to 200 keV capture cross-section evaluation was carried out with the help of FITACS code [Fr82] by fitting to data from [Wi83] and [We85]. Above 200 keV calculations were performed by statistical model. Fission cross-section evaluation above 42 keV was based on data

from [Se70], [Wi83], [Kn88] and [Be81]. The (n,2n) and (n,3n) reaction cross-section evaluation was carried out numerically by statistical model in accordance with [Ma76].

A.Blokhin and V.Maslov (PEI, Russia; EJI, Belorussia) carried out the BROND-2 evaluation. It was completed in 1990. The region of resolved resonances extends to 215 eV. Multilevel Breit-Wigner formula is used. Resonance parameters correspond to evaluation from [Ab81] and fission width correspond to JENDL-3 evaluation. The same evaluation was used for average resonance parameters in the region of unresolved resonances limited by the energy 30 keV from above. Above 30 keV capture cross-section was calculated by STAPRF code according to [Ma88]. In this case strength γ - function was chosen so that at 30 keV capture cross-section was equal to 2.2 barn. Fission cross-section at this energy range was evaluated from data of [Ka87], [Fu86], [Kn86] in accordance with [Ma88]. The (n,2n) and (n,3n) reaction cross-sections were calculated by statistical model according to [Ig88].

In Tables 7.22-7.28 some characteristics of interaction of neutrons with ^{243}Am nucleus obtained from various evaluations are compared among themselves and with recommended data.

Table 7.22

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{243}Am , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
σ_{f0}	0.894	0.116	0.074	0.092	0.07	1973, [5]
					$0.20^{+0.05}_{-0.05}$	1984, [6]
					$0.074^{+0.010}_{-0.010}$ *)	1993, [3]
$\sigma_{\gamma0}$	74.62	78.5	75.0	76.91	$79.3^{+8.0}_{-8.0}$	1973, [5]
					$75.1^{+1.8}_{-1.8}$	1984, [6]
					73^{+2}_{-2} *)	1993, [3]
					77^{+3}_{-3} **)	1993, [3]

*) - thermal column measurements

**) - measurements on reactor spectra with
average neutron energy of 0.0253 eV.

Table 7.23

The values of I_f , I_γ for ^{243}Am , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	7.610	6.115	6.165	5.996	10^{+6}_{-6}	1981, [35]
					$8.45^{+0.45}_{-0.45}$ *)	1988, [36]
					9^{+1}_{-1} **)	1975, [37]
I_γ	1823	1825	1821	1725	1820^{+70}_{-70}	1984, [6]
					1825^{+80}_{-80} ***)	1974, [38]
					2257^{+46}_{-46} **)	1993, [3]
					2250^{+20}_{-20} ****)	1993, [3]

*) - differential measurement, $E_{\text{max}} = 2 \text{ MeV}$

**) - integral measurement (or by integral measurements in
case of I_γ) by activation method with Cd shield

***) - differential measurement, $E_{\max} = 250 \text{ MeV}$

****) - reactor measurements

Table 7.24

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{243}Am , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.124	0.041	0.026	2.200	0.125	0.650
JENDL-3	1.092	0.183	0.006	2.141	0.360	0.221
ENDF/B-6	1.138	0.144	0.001	2.104	0.040	0.100
BROND-2	1.093	0.183	0.006	2.130	0.440	0.038
measur.	$1.145^{+0.023}_{-0.023}$, [9]					

Table 7.25

The value of ΔI_f for ^{243}Am , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	1.917	1.531	1.395	1.345
10 10^2	1.140	0.511	0.423	0.569
10^2 10^3	0.400	0.153	0.224	0.156
10^3 10^4	0.254	0.042	0.069	0.042
10^4 10^5	0.089	0.013	0.030	0.018
10^5 10^6	0.318	0.228	0.277	0.223
10^6 10^7	3.488	3.636	3.748	3.643

Table 7.26

The value of ΔI_γ for ^{243}Am , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	1613	1591	1588	1499
10 10^2	167.3	164.1	164.4	156.0
10^2 10^3	36.60	49.90	49.20	49.80
10^3 10^4	4.400	13.70	13.20	13.70
10^4 10^5	0.997	5.012	4.751	4.991
10^5 10^6	0.274	1.188	0.933	1.185
10^6 10^7	0.032	0.100	0.100	0.100

Table 7.27

The value of $\langle \sigma_f \rangle$ for ^{243}Am , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	0.608	0.376	0.346	0.327
10 10^2	0.453	0.171	0.147	0.190
10^2 10^3	0.142	0.052	0.080	0.052
10^3 10^4	0.097	0.015	0.024	0.015
10^4 10^5	0.029	0.007	0.011	0.007
10^5 10^6	0.275	0.197	0.250	0.197
10^6 10^7	1.578	1.742	1.787	1.746

Table 7.28

The value of $\langle \sigma_{\gamma} \rangle$ for ^{243}Am , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	317.4	314.3	313.6	300.1
10 10^2	52.85	51.71	51.77	50.15
10^2 10^3	8.562	16.83	16.57	16.79
10^3 10^4	1.708	4.809	4.584	4.809
10^4 10^5	0.371	1.866	1.769	1.858
10^5 10^6	0.102	0.442	0.347	0.441
10^6 10^7	0.006	0.019	0.020	0.019

7.5 ^{244g}Am

Isomer of ^{244g}Am is a short-lived one with $T_{1/2} = 10.1$ hours. Due to β - decay it transforms into relatively stable ^{244}Cm nucleus. ^{244g}Am is produced by neutron radiation capture by ^{243}Am nuclei, thereat, of two possible channels, $^{243}\text{Am}(n, \gamma)^{244g}\text{Am}$ and $^{243}\text{Am}(n, \gamma)^{244m}\text{Am}$, the second one is more probable in thermal and resonance neutron flux according to [5,6]. Fission process is the main transmutation channel of ^{244g}Am in various energies neutron flux. Data about characteristics of interaction of neutrons with ^{244g}Am are practically absent in literature, only recommended σ_{f0} value was

published in [5] and preserved in more later recommendation of [6]. The evaluations of characteristics of interaction between neutrons and ^{244}gAm in a number of versions compared by us are given merely in JENDL-3.

This evaluation (MAT 3955) was carried out by T.Nakagawa (JAERI, Japan), and it was completed in 1988. In the region of resolved resonances the cross-sections are given pointwise. The (n,2n) and (n,3n) reaction cross-sections were determined by evaporation model. Below 0.07 eV the 1/v law with normalizing on 600 barn at thermal point was adopted for capture cross-section. Above 0.07 eV capture cross section was calculated by statistical model with the help of CASTHY code [Ig75]. In this case strength γ - function corresponded to the average $\bar{D} = 0.13$ eV level spacing, obtained from level density parameters and average radiation width $\bar{\Gamma}_{\gamma} = 0.05$ eV. For fission cross-section below 0.07 eV the 1/v law with normalizing on 2300 ± 300 barn at thermal point was also adopted. Above 0.07 eV fission cross-section was assumed the same as for ^{242}gAm in JENDL-3 (MAT 3952).

Some characteristics of interaction of neutrons with ^{244}gAm corresponded to JENDL-3 evaluation are given in Tables 7.29 and 7.30.

Table 6.29

The values of σ_{f0} , $\sigma_{\gamma0}$, I_f , I_γ , σ_f , σ_{2n} , σ_{3n} for ^{244g}Am , in barn.

Value	JENDL-3	Recommend. value	Year of public.	Value	JENDL-3
σ_{f0}	2300	2300^{+300}_{-300}	1973, [5]	$\bar{\sigma}_\gamma$	0.325
		2300^{+300}_{-300}	1984, [6]	$\bar{\sigma}_{2n}$	0.031
$\sigma_{\gamma0}$	600.0	-		σ_f	2.427
I_f	1257	-		σ_{2n}	0.338
I_γ	315.8	-		σ_{3n}	0.340
$\bar{\sigma}_f$	1.759	-			

Table 7.30

The values of ΔI_f , ΔI_γ , $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{244g}Am , in barn.

JENDL-3 evaluation						
	$E_{\min}$ (eV)	$E_{\max}$	ΔI_f	ΔI_γ	$\langle \sigma_f \rangle$	$\langle \sigma_\gamma \rangle$
	0.5	10	969.5	244.8	228.1	57.70
	10	10^2	190.8	47.90	66.98	16.75
	10^2	10^3	60.20	14.68	21.17	5.116
	10^3	10^4	19.50	4.580	6.962	1.623
	10^4	10^5	8.916	2.010	3.190	0.749
	10^5	10^6	4.284	1.530	1.975	0.730
	10^6	10^7	4.100	0.270	1.890	0.045

7.6 ^{244m}Am

Isomer of ^{244m}Am is a short - lived one. It may desintegrate into

two channels, transforming into ^{244}Cm by β -decay with $T_{1/2} = 26$ min with a probability 99.96% and produced under neutron capture by ^{243}Am nuclei. Because of small half-life transmutation processes of $^{244\text{m}}\text{Am}$ isomer can constitute only small competition to β - decay of this nucleus in neutron flux under reactor's conditions. Nevertheless it must be pointed, that fission is the main transmutation process of $^{244\text{m}}\text{Am}$ in neutron flux of all energies.

As for $^{244\text{g}}\text{Am}$, there are no data about measured characteristics of interaction of neutron with $^{244\text{m}}\text{Am}$ in literature; and among the versions of evaluated data compared by us such evaluations are given in JENDL-3 only.

This evaluation (MAT 3956) was carried out by T. Nakagawa (JAERI, Japan), and it was completed in 1988. In resonance region the cross-sections are given point by point. Below 0.07 eV capture cross-section follows to $1/v$ law with the normalizing on 400 barn at thermal point. Above 0.07 barn capture cross-section was calculated by CASTHY code [Ig75], in this case strength function for γ - quanta corresponded to the average $\bar{D}=0.13$ eV level distance and $\Gamma_{\gamma} = 0.05$ eV radiative width. Fission cross-section below 0.07 eV also allows to $1/v$ law with normalizing on 1600 ± 300 barn at thermal point, and above 0.07 eV JENDL-3 (MAT 3952) evaluation for $^{242\text{g}}\text{Am}$ was used. The (n,2n) and (n,3n) reaction cross-sections were calculated by evaporation model and agree with similar cross-sections for $^{244\text{g}}\text{Am}$.

In Tables 7.31 and 7.32 some characteristics of interaction of neutrons with $^{244\text{m}}\text{Am}$ corresponding to JENDL-3 evaluation are given.

Table 7.31

The values of σ_{f0} , $\sigma_{\gamma 0}$, I_f , I_γ , $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{244m}Am , in barn.

Value	JENDL-3	Recommend. value	Year of public.	Value	JENDL-3
σ_{f0}	1600	1600^{+300}_{-300}	1973, [5]	$\bar{\sigma}_\gamma$	0.358
		1600^{+300}_{-300}	1984, [6]	$\bar{\sigma}_{2n}$	0.031
$\sigma_{\gamma 0}$	400.0	-		σ_f	2.427
I_f	1257	-		σ_{2n}	0.338
I_γ	315.6	-		σ_{3n}	0.340
$\bar{\sigma}_f$	1.759				

Table 6.32

The values of ΔI_f , ΔI_γ , $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{244m}Am , in barn.

JENDL-3 evaluation						
	$E_{\min}$	$E_{\max}$	ΔI_f	ΔI_γ	$\langle \sigma_f \rangle$	$\langle \sigma_\gamma \rangle$
	0.5	10	969.5	244.8	228.1	57.70
	10	10^2	190.8	47.90	66.98	16.75
	10^2	10^3	60.20	14.68	21.17	5.118
	10^3	10^4	19.50	4.580	6.962	1.626
	10^4	10^5	8.916	2.000	3.190	0.730
	10^5	10^6	4.284	1.160	1.975	0.571
	10^6	10^7	4.100	0.460	1.890	0.086

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^{241}Am

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^{242}Am

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^{244g}Am , ^{244m}Am

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8. Isotopes of $^{242-249}\text{Cm}$.

8.1 ^{242}Cm .

Even-even ^{242}Cm nucleus is a short-lived one with a half-life $T_{1/2}=162.8$ days. It transforms into ^{238}Pu due to α -decay. ^{242}Cm finds application in α -radiation sources and in radioisotope energy sources. ^{241}Am is an initial material for ^{242}Cm production. $^{242}\text{Cm}(\alpha)$ $^{238}\text{Pu}(n,\gamma)$ $^{239}\text{Pu}(n,\gamma)$ $^{240}\text{Pu}(n,\gamma)$ $^{241}\text{Pu}(\beta^-)$ $^{241}\text{Am}(n,\gamma)$ $^{242}\text{Am}(\beta^-)$ ^{242}Cm closed transforming chain takes place. Radiation capture is a main transmutation process of ^{242}Cm in thermal and resonance neutron flux although this nucleus also disintegrates by thermal neutron. In fission and thermonuclear neutron flux ^{242}Cm nucleus disintegrates dominantly. The only measurement result of $\sigma_{\gamma 0}$ from reactor spectrum was published. For the same experiment conditions a result of I_{γ} measurement is also known. Two results of $\sigma_{\gamma 0}$ measurements from reactor spectra and one of I_f value calculated by evaluated resonance parameters were published. In spite of scanty information about characteristics of interaction of neutrons with ^{242}Cm nucleus, their evaluations are represented in all versions of evaluated data compared by us.

R. Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7885) evaluation. 1983 is a year of evaluation completion. The region of resolved resonances extends to 275 eV. Resonance positions and neutron widths correspond to data from [Ar77]. Radiative width is assumed to be the same for all levels and is equal to 0.040 eV. For lower levels fission widths correspond to [Al88]; for the rest levels fission width is assumed to be the same for all levels and is equal to 0.004 eV. The 3.45 eV level was adopted at negative energy. Its

parameters were chosen so as to reproduce the cross-sections proposed in [6] at thermal point. The region of unresolved resonances is limited by the energy 40 keV from above. There were used the following energy independent parameters: $R=9.093$ Fermi, $S_0 = 0.92 \cdot 10^{-4}$, $S_2=0.97 \cdot 10^{-4}$, $\bar{\Gamma}_\gamma = 0.04$ eV. S_1 , $\bar{\Gamma}_f$ and $\bar{D}$ were energy dependent ones. At the energy 1 keV the values of $S_1=3.04 \cdot 10^{-4}$, $\bar{\Gamma}_f=0.093$ eV, $\bar{D}=17.16$ eV were used. Above 400 keV capture cross-section was calculated by statistical model with the help of CASTHY code [Ig75] in terms of competitive processes. In this case the average radiative width of 0.040 meV and level spacing of 18 eV were used. From 40 keV to 1 MeV fission cross-section evaluation was based on measurement results of [Al88] and [Vo84]. Above 1 MeV JENDL-2 evaluation was used. It was based on the energy run of ^{244}Cm fission cross-section from [Ig77] and on fission cross-section data near 4 MeV from [Be78]. The (n,2n) and (n,3n) reaction cross-sections were calculated by evaporation model according to [Pe73].

F. Mann, R. Schenter, J. Benjamin, B. McCrosson, R. Howerton (HEDL, SRL, USA) are the authors of ENDF/B-6 evaluation. 1978 is a year of evaluation completion. The region of resolved resonances is limited by the energy 276 eV from above. Resonance parameters were obtained according to the data from [Ar77]. Parameters of the bound level were evaluated by integral cross-section measurements from [Ih72] and [Ha51] which, it must be noted, were determined with deficient accuracy. The region of unresolved resonances extends up to 10 keV. In evaluation of the average resonance parameters the same basis as in evaluation of resolved ones was used. Strength function for p-neutrons was taken in accordance with [Ly68]. Above 10 keV fission, capture and the (n,2n), (n,3n) reaction cross-sections were

calculated by statistical model according to [Ma77]. It must be noted at least that ENDF/B-6 evaluation on cross-sections agrees with ENDF/B-5 one.

BROND-2 (MAT 9611) evaluation was carried out by A.Klepatskii, V.Maslov, Yu.Porodzinskii, E.Sukhovitskii, L.Bakhanovich (IJE, Belorussia), and it was completed in 1987. The region of resolved resonances extends to 155 eV. Single-level Breit-Wigner formula is used. Experimental data published in [Ar77] and [Al88] were used for evaluation of resonance parameters. The region of unresolved resonances is limited by the energy 42 keV from above. Average resonance parameters were obtained from the parameters of resolved levels. The values of $S_0=0.925 \cdot 10^{-4}$, $S_1=2.95 \cdot 10^{-4}$, $\bar{\Gamma}_f=4.45$ MeV, $\bar{\Gamma}_\gamma=33.7$ MeV, $\bar{D}(l=0)=8$ eV, $R = 9.14$ Fermi were used. Above 42 keV fission cross-section evaluation was carried out according to the experimental data from [Vo84], [Ca76], [Wi79] and to theoretical calculations from [Gr83]. Capture and the (n,2n) and (n,3n) reaction cross-sections were calculated by statistical model.

In Tables 8.1-8.7 the characteristics of interaction of neutrons with ^{242}Cm nucleus obtained by various evaluations are compared among themselves and with recommended data.

Table 8.1

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{242}Cm , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
σ_{f0}	5.011	5.064	3.025	5.008	< 5	1973, [5]
					< 5	1984, [6]
					4.6 ^{*)}	1972, [33]
$\sigma_{\gamma0}$	16.00	15.90	16.90	16.66	16 ⁺ ₅	1973, [5]
					16 ⁺ ₅	1984, [6]
					20 ⁺ ₅ ^{*)}	1976, [39]

*) - measurement on reactor spectrum

Table 8.2

The values of I_f , I_γ for ^{242}Cm , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	34.74	17.95	4.438	19.71	30	1970, [21]
					12.9 ⁺ _{0.7}	1988, [40]
I_γ	164.4	108.5	111.8	122.4	150 ⁺ ₄₀	1973, [5]
					110 ⁺ ₂₀	1984, [6]
					115 ⁺ ₅₃ ^{**))}	1977, [41]
					150 ⁺ ₄₀ ^{***))}	1969, [42]

*) - differential evaluation, $E_{\text{max}} = \text{keV}$

**) - differential evaluation

***) - measurements on reactor spectrum

Table 8.3

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{242}Cm , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.434	0.095	0.015	2.650	0.100	0.333
JENDL-3	1.840	0.070	0.003	2.907	0.101	0.007
ENDF/B-6	1.030	0.024	$3 \cdot 10^{-4}$	2.615	0.010	0.002
BROND-2	1.728	0.108	0.002	2.471	0.294	-

Table 8.4

The values of ΔI_f for ^{242}Cm , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	27.03	1.238	0.476	0.809
10	10^2	0.972	5.355	0.004	6.792
10^2	10^3	1.276	2.623	$1 \cdot 10^{-5}$	3.481
10^3	10^4	0.250	1.298	0	1.477
10^4	10^5	0.177	0.687	0.013	0.641
10^5	10^6	1.046	1.784	0.136	1.803
10^6	10^7	3.989	4.969	3.809	4.706

Table 8.5

The value of ΔI_{γ} for ^{242}Cm , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0&5	10	136.0	2.986	2.939	2.889
10	10^2	18.02	80.07	84.34	77.68
10^2	10^3	5.530	18.81	19.11	30.55
10^3	10^4	2.760	4.560	4.510	8.090
10^4	10^5	1.525	1.570	0.704	2.790
10^5	10^6	0.550	0.470	0.170	0.683
10^6	10^7	0.110	0.060	0.020	0.110

Table 8.6

The value of $\langle \sigma_f \rangle$ for ^{242}Cm , in barn.

$E_{\min}$	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	20.79	0.248	0.067	0.119
10	10^2	0.446	2.188	$7 \cdot 10^{-4}$	2.830
10^2	10^3	0.448	1.086	$2 \cdot 10^{-6}$	1.409
10^3	10^4	0.103	0.507	0	0.563
10^4	10^5	0.085	0.268	0.005	0.258
10^5	10^6	0.750	1.174	0.116	1.122
10^6	10^7	1.758	2.219	2.021	2.134

Table 8.7

The value of $\langle \sigma_{\gamma} \rangle$ for ^{242}Cm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	26.19	0.521	0.502	0.487
10 10^2	6.130	29.12	30.78	27.97
10^2 10^3	2.001	6.582	6.603	11.38
10^3 10^4	1.126	1.587	1.541	2.743
10^4 10^5	0.544	0.560	0.251	0.862
10^5 10^6	0.165	0.141	0.051	0.205
10^6 10^7	0.026	0.012	0.004	0.022

8.2 ^{243}Cm .

Even-odd ^{243}Cm nucleus is a relatively stable one, with $T_{1/2}=29.1$ years. It transforms into ^{239}Pu due to α -decay. ^{243}Cm is produced by neutron radiation capture by ^{242}Cm nucleus. Closed transmutation chain, $^{243}\text{Cm}(\alpha) \rightarrow ^{239}\text{Pu}(n, \gamma) \rightarrow ^{240}\text{Pu}(n, \gamma) \rightarrow ^{241}\text{Pu}(\beta^-) \rightarrow ^{241}\text{Am}(n, \gamma) \rightarrow ^{242}\text{Am}(\beta^-) \rightarrow ^{242}\text{Cm}(n, \gamma) \rightarrow ^{243}\text{Cm}$, takes place. ^{243}Cm disintegrates by neutrons of all energies, and an estimated fission process is the main transmutation channel of ^{243}Cm in neutron flux of all energies.

Few results of $\sigma_{\gamma 0}$ measurement by integral methods are known, and they, in average, agree with the value proposed in [6]. The σ_{f0} values obtained by the same methods are also agree with the recommendation from [6]. The discrepancies between I_f values obtained by integral methods by various authors account (according to [3]), approximately, for 8%; and the corresponding values agree with the proposed ones from [6]. Only one result of I_{γ} measurement on reactor, which agree both with recommendation from [25] and with more late one

from [6], is known.

The evaluations of interaction of neutrons with ^{243}Cm nucleus are represented in a number of compared by us versions of evaluated data in ENDL-82, JENDL-3, ENDF/B-6 and BROND-2.

R. Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7886) evaluation. 1983 is a year of evaluation completion.

JENDL-3 (MAT 3963) evaluation was carried out by T. Nakagawa (JAERI, Japan). It was completed in 1989. The region of resolved resonances extends to 70 eV. Breit-Wigner formula of isolated level is used. Resonance positions and neutron widths correspond to the data from [An82]. Radiative width is assumed to be the same (0.040 eV) for all resonances. Fission width for each level was evaluated as a difference between total width and a sum of neutron and radiative width. Total widths were taken from [An82]. In this case scattering radius was assumed to be equal to 10 Fermi. There was adopted a resonance at negative energy for reproduction of cross-sections proposed in [6] at thermal point. The region of unresolved resonances is limited by the energy 40 keV from above. Average resonance parameters were chosen with the help of ASREP code (Y.Kikuchi, Private Communication) so as to reproduce fission cross-section corresponding to data from [Si76], and total cross-section calculated by optical model. Energy independent parameters of $R=9.810$ Fermi, $S_2 = 1.7 \cdot 10^{-4}$, $\bar{\Gamma}_\gamma = 0.04$ eV, $\bar{\Gamma}_f = 1.481$ eV were used. S_0 , S_1 and $\bar{D}$ were the energy dependent parameters. At the energy of 1 keV, for example, they took the following values: $S_0 = 1.32 \cdot 10^{-4}$, $S_1 = 1.06 \cdot 10^{-4}$, $\bar{D} = 0.799$ eV. At the energies above 40 keV, capture cross-section was calculated by statistical model with the help of CASTHY code [Ig75]. The competitive fission and the (n,2n), (n,3n) processes were accounted. There were

also used the values of $\bar{\Gamma}_{\gamma}=0.040$ eV and $\bar{D}=0.809$ eV. Above 10 keV fission cross-section correspond to JENDL-2 evaluation based on data from [Si76]; from 10 keV to 3 MeV fission cross-section evaluation is based on data from [Fo87], and above 3 MeV independent evaluation was carried out. The (n,2n) and (n,3n) reaction cross-section were calculated by evaporation model in accordance with [Pl73].

ENDF/B-6 (MAT 9634) evaluation agrees with ENDF/B-5 one. The last evaluation was carried out by F.Mann, R.Schenter, J.Benjamin, R.Howerton (HEDL, SRL, LNL, USA). It was completed in 1978. The region of resolved resonances is limited by the energy 27 eV. Parameters of fifteen resolved resonances were taken from [Be72]. There was adopted a bound level, parameters of which were fitted for reproduction of cross-sections at thermal point from [Hu57]. The region of unresolved resonances extends up to 10 keV. Average resonance parameters were obtained by averaging of those of resolved ones from [Be72]. Capture cross-section at $E_n > 10$ keV and the (n,2n) and (n,3n) reaction cross-sections were calculated by statistical model in accordance with [Ma77]. Fission cross-section was calculated by the same model in the same energy range; measurement results agree with data from [Fu76].

In Tables 8.8-8.14 some characteristics of interaction of neutrons with ^{243}Cm nucleus obtained from various evaluations are represented. They are compared with available recommended and evaluated data.

Table 8.8

The values of σ_{f0} , $\sigma_{\gamma 0}$ for ^{243}Cm , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
σ_{f0}	690.8	617.4	692.5	$600^{+}_{-}50$	1973, [5]
				$617^{+}_{-}10$	1984, [6]
				$609^{+}_{-}26^{*)}$	1993, [3]
				$646^{+}_{-}37^{**)}$	1993, [3]
$\sigma_{\gamma 0}$	391.1	130.2	58.12	$225^{+}_{-}100$	1973, [5]
				$130^{+}_{-}10$	1984, [6]
				$130.7^{+}_{-}9.6^{*)}$	1977, [43]
				$137.4^{+}_{-}9.6^{**)}$	1977, [43]

*) - measurement (or measurements in case of σ_{f0}) on thermal column

**) - measurement (or measurements in case of $\sigma_{\gamma 0}$) on reactor spectrum with $E = 0.0253$ eV.

Table 8.9

The values of I_f , I_γ for ^{243}Cm , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
I_f	775.8	1559	1964	$1860^{+}_{-}400$	1973, [5]
				$1570^{+}_{-}100$	1984, [6]
				$1480^{+}_{-}150^{*)}$	1979, [44]
				$1606^{+}_{-}87^{**)}$	1993, [3]
I_γ	121.4	198.7	251.1	200	1972, [25]
				$215^{+}_{-}20$	1984, [6]
				$214.4^{+}_{-}20.3^{**)}$	1977, [43]

*) - activation measurements with Cd - shield

**) - reactor measurements

Table 8.10

The value of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{243}Cm , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.997	0.041	0.027	2.170	0.250	0.425
JENDL-3	1.938	0.091	0.022	2.500	0.451	0.075
ENDF/B-6	2.073	0.015	0.016	2.513	0.070	0.100

Table 8.11

The value of ΔI_f for ^{243}Cm , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6
0.5	10	562.6	1299	1615
10	10^2	133.5	173.5	246.8
10^2	10^3	50.60	50.90	68.20
10^3	10^4	13.35	18.90	18.70
10^4	10^5	6.890	7.600	5.907
10^5	10^6	4.458	4.800	4.381
10^6	10^7	4.650	4.300	4.800

Table 8.12

The value of ΔI_{γ} for ^{243}Cm , in barn.

$E_{\min}$	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6
0.5	10	90.38	152.4	190.7
10	10^2	17.07	31.02	45.45
10^2	10^3	7.400	10.38	11.27
10^3	10^4	5.030	3.370	2.920
10^4	10^5	1.120	1.152	0.770
10^5	10^6	0.449	0.620	0.140
10^6	10^7	0.030	0.120	0.016

Table 8.13

The value of $\langle \sigma_f \rangle$ for ^{243}Cm , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6
0.5	10	187.4	409.6	493.5
10	10^2	62.14	61.85	82.86
10^2	10^3	16.90	18.20	23.41
10^3	10^4	4.922	6.749	6.290
10^4	10^5	2.546	2.905	2.258
10^5	10^6	1.895	2.040	1.862
10^6	10^7	2.010	1.886	2.075

Table 8.14

The value of $\langle \sigma_{\gamma} \rangle$ for ^{243}Cm , in barn.

$E_{\min}$ (eV)	$E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6
0.5	10	27.52	47.00	60.45
10	10^2	6.096	10.35	14.68
10^2	10^3	3.068	3.644	3.832
10^3	10^4	1.875	1.162	0.967
10^4	10^5	0.384	0.395	0.263
10^5	10^6	0.100	0.138	0.035
10^6	10^7	0.007	0.023	0.003

8.3 ^{244}Cm .

Even-even ^{244}Cm is relatively long-lived nucleus ($T_{1/2}=18.1$ years) transforming into ^{240}Pu due to α -decay. ^{244}Cm is produced neither at neutron radiation capture by ^{243}Cm , or during β^- -decay of $^{244g,m}\text{Am}$ isomers. Following closed transmutation chains are possible:

transmutation channel of ^{244}Cm in thermal and resonance neutron flux

leading to production of ^{245}Cm nucleus. In contrast to ^{244}Cm , ^{245}Cm

nucleus disintegrates dominantly in flux of neutrons of all energies.

During ^{244}Cm burning out it is appropriate to place corresponding

samples in reactor's ranges where neutrons with the energies

0.5 eV $< E_n < 10$ eV are dominant in neutron spectrum. An estimated in

fission and thermonuclear neutron flux ^{244}Cm nucleus disintegrates

dominantly.

Some author's results of $\sigma_{\gamma 0}$ measurements by integral methods are known. They agree between themselves within the limits of assumed accuracies and with proposed values from [5,6]. Published results of few I_{γ} integral measurements, to which the recommended values from [5,6] correspond, also agree between themselves. The results of σ_{f0} measurements on thermal columns are, in average, below than those from thermal reactor spectra and from [5]; but they agree with proposed value from [6]. The value of I_f proposed in [5], in fact, corresponds to reactor measurement result, and the value from [35] is summed to be too high as it exceeds both the I_f value from [5] and proposed value from [3], which is the average-weighted quantity according to activation measurements with Cd - shield. The evaluation of characteristics of interaction of neutrons with ^{244}Cm are given in all versions of evaluated data compared by us.

R. Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7887) evaluation. 1983 is a year of evaluation completion.

JENDL-3 (MAT 3964) evaluation was carried out by T. Nakagawa (JAERI, Japan), and it was completed in 1989. The region of resolved resonances is limited by the energy 70 eV from above, but there were adopted levels at the energies of 646.9 eV, 759.7 eV, 914 eV, 971.5 eV. Above 20 eV the evaluation of resonance parameters was based on data from [Mo71], thereat for the levels just mentioned identical neutron width of 0.2 eV was assumed. Fission widths for the levels laying below are assumed in accordance with data from [Ma85]. Radiative width is assumed to be equal to 0.037 eV and scattering radius is equal to 11.2 Fermi (that correspond to scattering cross-section of 11.06 barn at 0.0253 eV in agreement with

recommendation from [6]). There was introduced a level at negative energy of 1.48 eV, and its parameters were adjusted to convey the recommended cross-sections at thermal point from [6]. Multilevel formula of Breit-Wigner type is used. The region of unresolved resonances is between the limits 70 eV and 40 keV. Average parameters were chosen with the help of ASREP code so as to reproduce fission cross-sections from [Ma85], total cross-section calculated by optical model and capture cross-section obtained by statistical model. In this case the following energy independent values were used: $R=9.221$ Fermi, $S_0=0.9 \cdot 10^{-4}$, $S_2=0.92 \cdot 10^{-4}$, $\bar{\Gamma}_\gamma=0.04$ eV. The values of S_1 , $\bar{\Gamma}_f$ and $\bar{D}$ depended from energy. For example, at the energy of 1 keV these parameters took the following values: $S_1=3.06 \cdot 10^{-4}$, $\bar{\Gamma}_f=0.00244$ eV, $\bar{D}=11.98$ eV. Above 40 keV capture cross-section was calculated by statistical model with the help of CASTHY code [Ig75] with allowance for competitive multiple processes and the values of $\bar{\Gamma}_\gamma=0.037$ eV, $\bar{D}=12$ eV. From 40 to 100 keV energy run of fission cross-section was evaluated by data of [Ma85]; from 100 to 800 keV JENDL-2 was preserved obtained under the fitting of semi-empirical formula to experimental results from [Mo71]; from 0.8 to 8 MeV experimental data from the works of the same authors and data from [Fo80] and [Vo84] were accounted at evaluation, above 8 MeV JENDL-2 evaluation was preserved. The (n,2n) and (n,3n) reaction cross-sections were calculated by evaporation model in accordance with [Pe65].

ENDF/B-6 (MAT 9637) evaluation agrees with ENDF/B-5 one. A.Mann, R.Schenter, R.Benjamin, R.Howerton, C.Dunfort, M.Alter (HEDL, SRL, LNL, ANC, AI, USA) are the authors of the last evaluation. 1978 is a year of evaluation completion. The region of resolved resonances is limited by the energy 525 eV from above. The parameters of 38

resonances were chosen in accordance with the measurement results from [Si71] and [Be72]. Parameters of the bound level and the first lowermost ones were modified according to reasonable experimental limits providing an agreement with integral data from [Be72], [To71] and experimental results from [Be75] and [Be77]. The region of unresolved resonances extends to 10 keV. Average parameters of s-resonances were obtained by averaging of resolved level parameters with the use of fission width from fission resonance measurements according to [Mo71]. For p - resonances the values of $\bar{D}$, $\bar{\Gamma}_\gamma$ and $\bar{\Gamma}_f$ were obtained by the same way; and the value of average neutron width was obtained with the help of the strength function for p - neutrons evaluated according to the data from [Du67]. Above 10 keV capture cross-section was calculated by statistical model described in [Ma77]. The (n,2n) and (n,3n) and fission reaction cross-sections of high energies were also calculated by the same model. At more lower energies up to 10 keV fission cross-section was evaluated from data of [Mo71].

A.Klepatskii, V.Maslov, Yu.Porodzinskii, E.Sukhovitskii, L.Bakhanovich A.Kolesov are the authors of BROND-2 (MAT 9631) evaluation. 1988 is a year of evaluation completion. The region of resolved resonances extends to 500 eV. Single-level Breit-Wigner formula is used. Resonance parameters were evaluated according to experimental data from [Co64], [Si72], [Mo71], [Be73] and [Ma85]. The region of unresolved resonances is limited by the energy 42.9 eV from above. Average resonance parameters were obtained by averaging of those of resolved levels. The following values were used: $S_0 = 1.218 \cdot 10^{-4}$ at the energy of 0.5 keV; $S_0 = 1.103 \cdot 10^{-4}$ at the energy of 43 keV; $S_1 = 3.3 \cdot 10^{-4}$ for all energies; $\bar{D}(l=0) = 11$ eV and

$\bar{\Gamma}_f(l=0)=1.35$ MeV at the energy 0.5 keV, $R = 9.5$ Fermi. For the energies above 42.9 keV the cross-section was calculated by statistical model as well as the (n,2n) and (n,3n) reaction cross-section. Fission cross-section evaluation at the same energy range used both the experimental data from [Fo80], [Vo84], [Ko68] and theoretical data from [Gr83].

In Tables 8.15-8.21 the comparison of some characteristics of interaction of neutrons with ^{244}Cm nucleus, obtained from various versions of evaluated data, with recommended data are given.

Table 8.15

The values of σ_{f0} , $\sigma_{\gamma 0}$ for ^{244}Cm , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend.	Year of
σ_{f0}	1.102	1.037	0.605	1.030	$1.2^{+0.1}_{-}$	1973, [5]
					$1.04^{+0.20}_{-}$	1984, [6]
					$1.02^{+0.5*}_{-}$	1993, [3]
					$1.5^{+1.0**}_{-}$	1971, [45]
$\sigma_{\gamma 0}$	10.98	15.10	10.38	15.26	$13.9^{+1.0}_{-}$	1973, [5]
					$15.2^{+1.2}_{-}$	1984, [6]
					$15.2^{+1.2*}_{-}$	1968, [46]
					$13.1^{+2.2**}_{-}$	1993, [3]

*) - measurements (or measurement for $\sigma_{\gamma 0}$) on
thermsl columns

**) - measurements (or measurement for $\sigma_{\gamma 0}$) on
thermal reactor spectrum

Table 8.16

The values of I_f , I_γ for ^{244}Cm , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	BROND-2	Recommend. value	Year of public.
I_f	33.44	11.52	16.68	11.38	$12.5^{+2.5}_{-2.5}$	1973, [5]
					18^{+2}_{-2}	1981, [35]
					$14.1^{+1.4}_{-1.4}^{*)}$	1993, [3]
					$12.5^{+2.5}_{-2.5}^{**})$	1971, [45]
I_γ	611.9	660.5	595.6	636.1	650^{+50}_{-50}	1973, [5]
					650^{+30}_{-30}	1984, [6]
					$643^{+53}_{-53}^{***})$	1975, [47]
					$626^{+53}_{-53}^{*)}$	1968, [46]
					$650^{+25}_{-25}^{**})$	1993, [3]

*) - activation measurements (or measurement in case of I_γ)
with Cd - shield

**) - measurements (or measurement in case of I_γ)
on reactor

***) - differential evaluation

Table 8.17

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{244}Cm , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.400	0.042	0.015	2.650	0.100	0.333
JENDL-3	1.572	0.139	0.007	2.636	0.310	0.091
ENDF/B-6	1.616	0.119	0.006	3.080	0.065	0.470
BROND-2	1.585	0.122	0.004	2.322	0.368	0.055

Table 8.18

The value of ΔI_f for ^{244}Cm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	25.82	1.961	7.188	2.172
10 10^2	0.976	2.783	2.607	2.694
10^2 10^3	1.273	0.647	0.671	0.616
10^3 10^4	0.249	0.421	0.281	0.145
10^4 10^5	0.181	0.150	0.153	0.138 [*]
10^5 10^6	1.037	1.055	1.063	1.045
10^6 10^7	3.901	4.471	4.717	4.591

Table 8.18

The value of ΔI_γ for ^{244}Cm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5 10	514.3	571.4	507.5	553.2
10 10^2	62.36	57.48	53.93	55.45
10^2 10^3	28.38	23.05	22.88	21.50
10^3 10^4	5.240	5.830	7.580	3.670
10^4 10^5	0.910	1.567	2.459	1.150
10^5 10^6	0.690	1.031	1.171	0.950
10^6 10^7	0.020	0.160	0.080	0.140

Table 8.20

The value of $\langle \sigma_f \rangle$ for ^{244}Cm , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	20.585	1.498	5.722	1.568
10	10^2	0.449	0.998	0.915	0.928
10^2	10^3	0.446	0.238	0.263	0.219
10^3	10^4	0.103	0.153	0.104	0.053
10^4	10^5	0.085	0.059	0.060	0.054
10^5	10^6	0.744	0.835	0.841	0.816
10^6	10^7	1.731	2.031	2.198	2.104

Table 8.21

The value of $\langle \sigma_\gamma \rangle$ for ^{244}Cm , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6	BROND-2
0.5	10	409.81	457.2	406.4	442.1
10	10^2	25.26	22.78	21.99	22.00
10^2	10^3	11.14	8.759	8.824	7.605
10^3	10^4	1.661	2.029	2.921	1.258
10^4	10^5	0.421	0.750	1.044	0.598
10^5	10^6	0.102	0.243	0.276	0.224
10^6	10^7	0.007	0.030	0.017	0.030

8.4 ^{245}Cm .

Even-even ^{245}Cm is a practically stable nucleus ($T_{1/2} = 8.5 \cdot 10^3$ years, α - decay); it is produced under neutron capture by ^{244}Cm . An estimated the fission process is the main transmutation process of

^{245}Cm nucleus in flux of neutrons of all energies. There were published some works where the results of experimental evaluation of $\bar{\sigma}_{f0}$, $\bar{\sigma}_{\gamma 0}$, $\bar{I}_f$ and $\bar{I}_{\gamma}$ value measured by various methods are represented. These results are in good agreement both among themselves and with the values proposed in [5,6]. Among the evaluated data versions compared by us the evaluations of characteristics of interaction between neutrons and ^{245}Cm nucleus are given in ENDL-82, JENDL-3 and ENDF/B-6.

R. Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7898) evaluation. It was completed by him in 1983. JENDL-3 (MAT 3965) evaluation was carried out by T. Nakagawa (JAERI, Japan). It was completed in 1989. The region of resolved resonances is limited by the energy 60 eV from above. Breit-Wigner formula for isolated levels is used. Reich-Moore parameters at the energies higher than 20 eV were taken from [Mo71], and at the energies below 20 eV they were taken from [Br78] with a small modification of level parameters at negative energy for a greater cross-section agreement to experimental data at thermal point. To account the differences between Reich-Moore and Breit-Wigner formulas for isolated level background cross-section was introduced. The 0.04 meV radiative width and scattering radius of 10 Fermi were used. The region of unresolved resonances is limited by the energy 40 keV from above. Average resonance parameters were chosen with the help of ASREP code so as to reproduce fission cross-section evaluation from [Mo71], total cross-section calculated by optical model and capture cross-section calculated by statistical model. Following energy independent parameters were used: $R=9.43$ Fermi, $S_0=1.02 \cdot 10^{-4}$, $S_1=2.24 \cdot 10^{-4}$, $S_2=0.9 \cdot 10^{-4}$, $\bar{\Gamma}_{\gamma}=0.04$ eV. $\bar{\Gamma}_f$ and $\bar{D}$ depended from energy. For example, at the energy of 1 keV they took

the values: $\bar{\Gamma}_f = 2.01$ eV, $\bar{D} = 1.397$ eV. Above 40 keV capture cross-section was calculated by statistical model with the help of CASTHY code [Ig75] taking into account of competitive processes. There were used the values of $\Gamma_\gamma = 0.040$ eV and $D = 1.4$ eV from [6]. The (n,2n) and (n,3n) reaction cross-section were calculated by evaporation model according to [Pe65]. At the energies below 100 keV JENDL-2 evaluation was preserved for fission cross-section. Fission cross-section was evaluated by semi-empirical formula fitted to experimental results from [Mo71]; above 100 keV fission cross-section evaluation was carried out on a base of experimental data set from [Wh83].

ENDF/B-6 (MAT 9640) evaluation agrees with ENDF/B-5 one, carried out by R. Benjamin and R. Howerton (SRL, LNLL, USA). It was completed in 1979. The region of resolved resonances is limited by the energy 60.5 eV. The 38 levels and one bound level were accounted. Their parameters correspond to the data from [Mo71] and [Br78]. Cross-sections at the thermal point are given pointwise. The region of unresolved resonances is limited by the energy 10 keV from above. Average parameters for s - resonances were obtained by averaging of appropriate parameters of resolved levels. Strength function for p-neutrons ($2.0 \cdot 10^{-4}$) was obtained by extrapolation over data from [Ly68]. Capture cross-section evaluations above 10 keV and those of the (n,2n), (n,3n) reaction agree with ENDL-78 [MO78]. Fission cross-section evaluation agree with ENDL-78 ones, except the energy range from 0.01 MeV to 0.1 MeV, where data from [Na78] were used.

In Tables 8.22-8.28 of some characteristics of interaction of neutrons with ^{245}Cm obtained from various versions of evaluated data with recommended values are given.

Table 8.22

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{245}Cm , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
σ_{f0}	2020	2001	2223	2020^{+40}_{-}	1973, [5]
				2145^{+58}_{-}	1984, [6]
				2143^{+58*}_{-}	1978, [48]
				2043^{+38**}_{-}	1993, [3]
				1980^{+32***}_{-}	1993, [3]
$\sigma_{\gamma0}$	391.1	346.4	342.6	345^{+20}_{-}	1973, [5]
				369^{+17}_{-}	1984, [6]
				341^{+34*}_{-}	1978, [48]
				336^{+16***}_{-}	1993, [3]

*) - differential measurement

**) - measurements on thermal columns

***) - measurements on thermal reactor spectrum

Table 8.23

The values of I_f , I_γ for ^{245}Cm , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
I_f	830.8	798.4	838.6	750^{+150}_{-}	1973, [5]
				840^{+40}_{-}	1984, [6]
				721^{+140*}_{-}	1978, [48]
				835^{+53**}_{-}	1993, [3]
				750^{+50***}_{-}	1968, [49]
I_γ	121.4	110.2	109.2	101^{+8}_{-}	1973, [5]
				101^{+8}_{-}	1984, [6]
				100^{*}_{-}	1978, [48]
				102^{+3***}_{-}	1993, [3]

*) - differential evaluation

**) - activation measurement with Cd - shield

***) - measurement (or measurements in case of I_γ)
on reactor

Table 8.24

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{245}Cm , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.723	0.041	0.028	2.550	0.250	0.425
JENDL-3	1.734	0.064	0.024	2.540	0.360	0.141
ENDF/B-6	1.829	0.041	0.028	2.170	0.250	0.425

Table 8.25

The value of ΔI_f for ^{245}Cm , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6
0.5	10	602.1	599.3	609.0
10	10^2	144.4	120.8	139.1
10^2	10^3	51.66	47.71	58.21
10^3	10^4	17.51	15.58	18.01
10^4	10^5	6.300	6.092	5.395
10^5	6			
10^6	10^7	4.100	4.080	4.600

Table 8.26

The value of ΔI_γ for ^{245}Cm , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6
0.5	10	90.38	75.05	76.28
10	10^2	17.07	22.88	21.84
10^2	10^3	7.410	8.477	7.325
10^3	10^4	5.030	2.260	2.190
10^4	10^5	1.210	1.230	1.210
10^5	10^6	0.280	0.290	0.280
10^6	10^7	0.030	0.040	0.030

Table 8.27

The value of $\langle \sigma_f \rangle$ for ^{245}Cm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6
0.5 10	210.0	204.8	205.8
10 10^2	65.08	52.45	60.70
10^2 10^3	17.74	16.71	20.26
10^3 10^4	6.249	5.547	6.308
10^4 10^5	2.611	2.604	2.306
10^5 10^6	1.706	1.745	1.370
10^6 10^7	1.877	1.852	2.004

Table 8.28

The value of $\langle \sigma_\gamma \rangle$ for ^{245}Cm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6
0.5 10	27.53	27.10	27.22
10 10^2	6.097	9.969	8.860
10^2 10^3	3.068	2.851	2.537
10^3 10^4	1.875	0.797	0.762
10^4 10^5	0.384	0.420	0.384
10^5 10^6	0.100	0.152	0.100
10^6 10^7	0.007	0.008	0.007

8.5 ^{246}Cm .

Even-even ^{246}Cm is a practically stable nucleus ($T_{1/2} = 4.73 \cdot 10^3$ years, α -decay). It is produced under neutron capture by ^{245}Cm

nucleus. An estimated in thermal and resonance neutron flux ^{246}Cm nucleus capture neutrons dominantly with production of also practically stable ^{247}Cm nucleus, which probability of disintegration by thermal and resonance neutrons already exceeds capture probability. In fission and thermonuclear neutron flux ^{246}Cm nucleus disintegrates dominantly. Few results of $\sigma_{\gamma 0}$ measurements by integral methods are known. The result from [5] exceeding by an order both the data from other works and proposed values from [5,6] stands out among them. According to author's recommendation from [3] let us consider the $\sigma_{\gamma 0}$ measurement result from [50] to be untrue. The results of I_{γ} evaluation by integral methods agree among themselves and with recommended values from [5,6]. The results of $\sigma_{\gamma 0}$ and I_f measurements are represented more poorly, but available data in generally agree among themselves and with recommended values from [5,6].

Among neutron data versions compared by us the evaluations of characteristics of interactions between neutrons and ^{246}Cm are represented in ENDL-82, JENDL-3 and ENDF/B-6. R.Howerton (LNLL, USA) is the author of ENDL-82 evaluation. 1987 is a year of evaluation completion. JENDL-3 (MAT 3966) evaluation was carried out by Y.Kikuchi, T.Nakagawa (JAERI, Japan). It was completed in 1987. The region of resolved resonances is limited by the energy 330 eV. Multilevel Breit-Wigner formula is used. The following data were used in evaluation of resonance parameters: [Mo71], [Be74] in evaluation of resonance positions; [Be72], [Be74], [Be75] in evaluation of neutron widths; [be74], [Mo71] in evaluation of the average radiative width of 0.031 meV; [Mo71], [St82], [Ma85] in evaluation of fission widths; level fission width at the energy 4.315 eV was chosen for cross-section description at thermal point; scattering radius of 9.85

Fermi was chosen to conserve scattering cross-section of 11.1 barn proposed in [6] at thermal point. Backing cross-section followed the $1/V$ law was introduced. The region of unresolved resonances extends to 30 keV. The values of $S_0 = 0.94 \cdot 10^{-4}$, $S_1 = 3.17 \cdot 10^{-4}$, $S_2 = 0.88 \cdot 10^{-4}$, $R = 9.15$ Fermi were obtained by spherical optical model. The values of $\bar{D} = 31.7$ eV and $\Gamma_\gamma = 31$ MeV were obtained by averaging of resolved level parameters. Average fission width was obtained at fitting to data from [St82]. At the energies above 30 keV capture cross-section was calculated by statistical model with the help of CASTHY code [Ig75] in terms of competitive processes, and $9.76 \cdot 10^{-4}$ strength function of γ -quanta was obtained from resonance parameters. Fission cross-section evaluation at the same energy range was based on the results from [St82], [Fo80]. The (n,2n) and (n,3n) reaction cross-sections were calculated by evaporation model in accordance with data from [Pe65]. ENDF/B-6 (MAT 9643) evaluation agrees with ENDF/B-5 one carried out by author's group from BNL. SRL and LNLL were completed in 1976. The region of resolved resonances is limited by the energy 385 eV from above. Parameters of 10 s - resonances were chosen according to the data from [Be74], [Be72], [Mo71] and permit, within the limits of two standard deviations, to be in agreement with experimental data from [Be72b], [Be75], [Ma69]. Potential scattering cross-section of 10.466 barn was obtained by optical model. The region of unresolved resonances extends to 10 keV. Average parameters of s-resonances were determined according to data from [Mo71]. Average parameters of p-resonances were chosen so as to have smooth transition to capture cross-section near 10 keV evaluated by calculations in the region of smooth cross-sections. Above 10 keV capture cross-section was calculated by optical model with the help of COMNUC code [Du71] with

smooth transition from 10 MeV to LNLL evaluation [Mo76]. In fission cross-section evaluation above 10 keV data from [Mo71] were used. The (n,2n) and (n,3n) reaction cross-sections correspond to LNLL evaluation [Ho76].

In Tables 8.29-8.35 the comparison of some characteristics of interaction between neutrons and ^{246}Cm nucleus obtained from various versions of evaluated data with recommended data is given.

Table 8.29

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{246}Cm , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
σ_{f0}	0.170	0.140	0.063	$0.17^{+0.10}_{-0.10}$	1973, [5]
				$0.14^{+0.05}_{-0.05}$	1984, [6]
				$0.158^{+0.080}_{-0.080}^{*)}$	1993, [3]
$\sigma_{\gamma0}$	1.201	1.291	1.297	$1.3^{+0.3}_{-0.3}$	1973, [5]
				$1.22^{+0.16}_{-0.16}$	1984, [6]
				$1.16^{+0.3}_{-0.3}^{*)}$	1993, [3]
				$1.5^{+0.5}_{-0.5}^{**})$	1971, [45]

*) - thermal column measurements

**) - thermal reactor measurements

Table 8.30

The values of I_f , I_γ for ^{246}Cm , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
I_f	6.252	8.243	9.020	$10.0_{-0.4}^{+0.4}$	1973, [5]
				$10.2_{-0.4}^{+0.4}$	1984, [6]
				$10.4_{-0.8}^{+0.8*})$	1993, [3]
I_γ	139.5	113.1	104.2	121_{-7}^{+7}	1973, [5]
				121_{-7}^{+7}	1984, [6]
				$101_{-11}^{+11**})$	1974, [51]
				$120_{-10}^{+10*})$	1993, [3]
				$128_{-1}^{+1***})$	1993, [3]

*) - activation measurements with Cd-shield

**) - differential evaluation

***) - reactor measurement

Table 8.31

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{246}Cm , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.347	0.041	0.024	2.100	0.175	0.700
JENDL-3	1.237	0.056	0.007	2.420	0.385	0.245
ENDF/B-6	1.384	0.042	0.008	2.100	0.035	0.803

Table 8.32

The value of ΔI_f for ^{246}Cm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6
0.5 10	0.654	2.989	3.757
10 10^2	0.537	0.293	0.358
10^2 10^3	0.265	0.315	0.174
10^3 10^4	0.107	0.211	0.021
10^4 10^5	0.117	0.092	0.001
10^5 10^6	0.600	0.442	0.463
10^6 10^7	3.972	3.901	4.246

Table 8.33

The value of ΔI_γ for ^{246}Cm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6
0.5 10	87.60	76.38	69.79
10 10^2	33.57	22.81	22.62
10^2 10^3	12.64	9.211	8.637
10^3 10^4	4.200	3.270	2.270
10^4 10^5	1.164	1.059	0.690
10^5 10^6	0.306	0.320	0.160
10^6 10^7	0.020	0.050	0.040

Table 8.34

The value of $\langle \sigma_f \rangle$ for ^{246}Cm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6
0.5 10	0.282	1.342	1.695
10 10^2	0.316	0.186	0.227
10^2 10^3	0.104	0.169	0.065
10^3 10^4	0.047	0.077	0.005
10^4 10^5	0.052	0.034	$5 \cdot 10^{-4}$
10^5 10^6	0.471	0.345	0.414
10^6 10^7	1.754	1.824	1.927

Table 8.35

The value of $\langle \sigma_\gamma \rangle$ for ^{246}Cm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6
0.5 10	39.66	34.47	31.48
10 10^2	20.82	15.11	14.48
10^2 10^3	4.310	3.814	3.416
10^3 10^4	1.638	1.166	0.744
10^4 10^5	0.376	0.349	0.221
10^5 10^6	0.102	0.109	0.087
10^6 10^7	0.007	0.010	0.008

8.6 ^{247}Cm .

Even-odd ^{247}Cm nucleus is a practically stable one ($T_{1/2} = 1.56 \cdot 10^7$ years, α - decay). It is produced under radiation neutron capture by ^{246}Cm nuclei. This reaction goes more intensively

in neutrons flux with the energies from about 0.5 eV to about 100 eV. An estimated in thermal and resonance neutron flux fission and capture reactions, with some domination of the first, are the main channels of ^{247}Cm transmutation. In fission and thermonuclear neutron flux ^{247}Cm nucleus disintegrated dominantly.

As was mentioned in [3], in not numerous data about the results of $\sigma_{\gamma 0}$ measurements the errors have reached about 50%. The same is the error of $\sigma_{\gamma 0}$ proposed value from [5], that is appreciably larger than the uncertainty of proposed value from [6]. The errors of I_{γ} values evaluated by some authors are rather large. In particular, recommended values of I_{γ} from [5] and [6] differ noticeably from each other, but they agree among themselves within the limits of errors. This occurs first and foremost because of 50% error interval for I_{γ} value proposed in [5]. The evaluations of σ_{f0} and I_f values have more high accuracy, thereat for the last characteristic evaluation from [52] is summed to be excessive. It was excluded from consideration in [3] when defining of average - weighted evaluation. The evaluations of characteristics of interaction between neutrons and ^{247}Cm are given in ENDL-82, JENDL-3 and ENDF/B-6.

R. Howerton (LNLL, USA) is the author of ENDL-82 (MAT 7890) evaluation. 1983 is a year of evaluation completion.

JENDL-3 (MAT 3967) was carried out by T. Nakagawa, Y. Kikuchi (JAERI, Japan). It was completed in 1989. The region of resolved resonances extends to 60 eV. Multilevel formula of Breit-Wigner is used. Experimental data from [Mo71] and [Be79] were used in evaluation of resonance parameters. Radiative widths took the value 0.040 meV and scattering radius was equal to 9.14 Fermi. Level parameters at the energy 1.25 eV were taken according to recommendation from [6]. The

0.3 eV level was introduced at negative energy. The region of unresolved resonances is limited by the energy 30 keV. Average resonance parameters were chosen with the help of ASREP code so as to reproduce fission cross-section evaluation from [Mo71], total cross-section calculated by optical model and capture cross-section obtained by statistical model with the help CASTHY code [Ig75]. The values of $R = 9.386$ Fermi, $S_2 = 0.86 \cdot 10^{-4}$, $\Gamma_\gamma = 0.040$ eV, $\bar{\Gamma}_f(J^\pi=4^-) = 0.0534$ eV, $\bar{\Gamma}_f(5^-) = 0.5$ eV, $\bar{\Gamma}_\gamma(3^+) = 0.08$ eV, $\bar{\Gamma}_f(4^+) = 0.68$ eV, $\bar{\Gamma}_f(5^+) = 0.05$ eV, $\bar{\Gamma}_f(6^+) = 0.47$ eV were energy independent. Fission widths were evaluated with the help of data from [Ki71]. The values of S_0 , S_1 and $\bar{D}$ were energy dependent. For example, at the energy 0.9 keV these parameters took the values: $S_0 = 0.77 \cdot 10^{-4}$, $S_1 = 2.89 \cdot 10^{-4}$, $\bar{D} = 1.397$ eV. At the energies above 30 keV capture cross-section was calculated by statistical model with the help of CASTHY code [Ig75] in terms of the competitive processes and with the $2.29 \cdot 10^{-2}$ value of γ -quanta evaluated by resonance parameters. The $(n,2n)$ and $(n,3n)$ reaction cross-sections were calculated by evaporation model according to [Pe65]. Below 50 keV fission cross-section evaluation was carried out by experimental data from [Mo71], and above this energy it was performed according to data from [Fo87].

ENDF/B-6 (MAT 9646) evaluation agrees with ENDF/B-5 one. The last is a combination of two evaluations performed by R. Benjamin, F. McCrosson (BNL, SRL, USA) and by R. Howerton (LNL, USA). It was completed in 1976. The region of resolved resonances extends to 61.7 eV and includes the characteristics of 35 resonances. Parameters of the first five resonances were chosen with the help of GENRPAR code [Mc71] so as to reproduce integral data from [Be72] and [Be75]. A

bound level was introduced to reproduce cross-sections at thermal point from [6]. Parameters of the rest 29 resonances were chosen with accordance to the data from [Mo71] obtained under nuclear explosions. The region of unresolved resonances is limited by the energy 10 keV from above. Average resonance parameters were obtained by averaging of resolved s-resonance parameters. Above 10 keV the LNLL evaluation [Ho76] was used for all parameters.

In Tables 8.36-8.42 some characteristics of interaction between neutrons and ^{247}Cm nucleus, obtained from various versions of evaluated data, are compared among themselves and with proposed or experimentally evaluated data.

Table 8.36

The values of σ_{f0} , $\sigma_{\gamma0}$ for ^{247}Cm , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
σ_{f0}	82.53	81.79	83.44	90_{-10}^{+10}	1973, [5]
				$81.9_{-4.4}^{+4.4}$	1984, [6]
				$106_{-53}^{+53}^{*)}$	1991, [53]
				$98.3_{-19.7}^{+19.7}^{**}$	1993, [3]
				$100_{-50}^{+50}^{***})$	1971, [45]
$\sigma_{\gamma0}$	90.09	57.20	58.17	60_{-30}^{+30}	1973, [5]
				57_{-10}^{+10}	1984, [6]
				$60_{-20}^{+20}^{**})$	1968, [46]
				$60_{-30}^{+30}^{***})$	1971, [45]

*) - differential evaluation

**) - measurements (or measurement in case of $\sigma_{\gamma0}$) on thermal column

***) - measurement on thermal reactor spectrum

Table 8.37

The values of I_f , I_γ for ^{247}Cm , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
I_f	917.3	610.9	751.6	880_{-100}^{+}	1973, [5]
				760_{-50}^{+}	1984, [6]
				$775_{-32}^{+*})$	1993, [3]
				$935_{-190}^{+**})$	1971, [45]
I_γ	364.1	535.6	493.2	800_{-400}^{+}	1973, [5]
				530_{-30}^{+}	1984, [6]
				$490_{-100}^{+*})$	1968, [46]
				$800_{-400}^{+**})$	1971, [45]

*) - activation measurements (or measurement in case
of I_γ) with Cd-shield

**) - reactor measurements

Table 8.38

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{247}Cm , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	2.066	0.041	0.041	2.500	0.100	0.500
JENDL-3	1.915	0.072	0.018	2.540	0.241	0.275
ENDF/B-6	2.066	0.041	0.041	2.500	0.100	0.500

Table 8.39

The value of ΔI_f for ^{247}Cm , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6
0.5	10	790.3	489.6	684.6
10	10^2	68.33	61.06	40.90
10^2	10^3	36.52	35.77	7.390
10^3	10^4	8.290	10.250	4.860
10^4	10^5	4.280	5.280	4.280
10^5	10^6	4.650	4.190	4.650
10^6	10^7	4.900	4.680	4.900

Table 8.40

The value of ΔI_γ for ^{247}Cm , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6
0.5	10	291.7	497.7	450.0
10	10^2	42.61	27.36	32.79
10^2	10^3	23.74	5.910	6.100
10^3	10^4	4.500	2.230	2.850
10^4	10^5	1.280	1.810	1.280
10^5	10^6	0.210	0.560	0.210
10^6	10^7	0.020	0.040	0.020

Table 8.41

The value of $\langle \sigma_f \rangle$ for ^{247}Cm , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6
0.5	10	108.3	102.7	98.34
10	10^2	36.57	33.10	16.33
10^2	10^3	13.00	12.95	2.611
10^3	10^4	3.037	3.658	3.437
10^4	10^5	1.926	2.050	1.926
10^5	10^6	1.953	1.869	1.953
10^6	10^7	2.164	2.159	2.164

Table 8.42

The value of $\langle \sigma \rangle$ for ^{247}Cm , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6
0.5	10	52.17	86.82	64.63
10	10^2	22.62	12.53	12.68
10^2	10^3	8.454	2.151	2.117
10^3	10^4	1.676	0.901	1.301
10^4	10^5	0.374	0.718	0.374
10^5	10^6	0.102	0.170	0.102
10^6	10^7	0.005	0.008	0.005

8.7 ^{248}Cm .

Even-even ^{248}Cm is a practically stable nucleus ($T_{1/2}=3.40 \cdot 10^5$ years, α -decay). It is produced at radiation neutron capture by ^{247}Cm .

This process is the most intensive in neutron flux with the energies from about 0.5 eV to ≈ 10 eV. Capture leading to production of short-lived ^{249}Cm nucleus is a main transmutation channel of ^{248}Cm in thermal and especially resonance neutron flux. In fission and thermonuclear neutron flux ^{248}Cm disintegrate dominantly. Data resulting from some works devoted to experimental evaluation of $\sigma_{\gamma 0}$ value for ^{248}Cm , except this characteristic evaluation at thermal column (data from [46]), are as a whole overlapped by each other, although the evaluation of average-weighted $\sigma_{\gamma 0}$ value according to measurement results at thermal reactor spectrum leads to slightly excessive evaluation value in comparison with other evaluation methods. The result of $\sigma_{\gamma 0}$ evaluation is obviously overestimated and it was excluded from consideration in [3] when determining average-weighted value in corresponding measurement group. The results of some authors devoted to experimental evaluation of I_{γ} value are in good agreement, but differential evaluation was obtained only for $E_{\text{max}}=3$ keV. There are known only two measurements of σ_{f0} value for ^{248}Cm at thermal columns, which are in good agreement with themselves and with proposed values from [5,6]. The results of activation measurements of I_f agree with recommendation from [5], but they are noticeably less than the I_f value proposed for ^{248}Cm in [6].

The evaluations of characteristics of interaction between neutrons and ^{248}Cm nucleus are given in ENDL-82, JENDL-3 and ENDF/B-6.

R. Howerton (LNLL, USA) is the authors of ENDL-82 (MAT 7891) evaluation. 1982 is a year of evaluation completion.

JENDL-3 (MAT 3968) evaluation was carried out by Y. Kikuchi, T. Nakagawa (JAERI, Japan). It was completed in 1984. The region of resolved resonances extends to 1.5 keV. Resonance positions, neutron

and radiative widths were obtained from experimental data of [Be74]. According to the data from this work for resonances, whose radiative widths were not determined by experimental way, this characteristic was assumed to be equal to 0.026 eV. Fission widths were evaluated according to [Mo71] and the average 0.0013 fission width was assumed. Last quantity was used as a characteristic for resonances for which fission widths could not be determined from available experimental data. Scattering radius was assumed to be equal to 9.1 Fermi for reproduction of potential scattering cross-section of 10.4 barn adopted in [Be74]. Neutron width for the first resonance was slightly corrected to reproduce capture cross-section of 2.57 barn at thermal point. Backing cross-sections were adopted only for total and fission cross-sections and they followed the $1/V$ law. Cross-sections at thermal point reproduce the evaluations following from experimental data. The region of unresolved resonances is limited by the energy 30 keV from above. The values of $S_1 = 3.32 \cdot 10^{-4}$, $S_2 = 0.844 \cdot 10^{-4}$, $R=8.88$ Fermi were obtained in optical-model calculations. The values of $\bar{D}=400$ eV, $\bar{\Gamma}_\gamma = 26$ MeV, $S_0 = 1.2 \cdot 10^{-4}$ were evaluated from parameters of resolved resonances. Average fission width was obtained by fitting to data from [St82]. Above 30 keV capture cross-section was calculated by statistical model with the help of CASTHY code [Ig75] in terms of competitive processes and with strength function of γ -quanta of $6.5 \cdot 10^{-4}$ evaluated from resonance parameters. The (n,2n) and (n,3n) reaction cross-sections were calculated by evaporation model according to [Pe65]. Fission cross-section was evaluated on a basis of experimental data from [St82] and [Fo80].

• ENDF/B-6 (MAT 9649) agrees with previous ENDF/B-5 one. The last was carried out by F.Mann, R.Schenter, R. Benjamin and R.Howerton

(HEDL, SRL, LNLL, USA). It was completed in 1978. The region of resolved resonances is limited by the energy 2.4 keV from above and includes 46 s-resonances according to [Be74], [Ke71]. The evaluation of average fission width was carried out by integral data from [Be72] and [Dr73]. Bound level is missing. Capture cross-section at thermal point corresponds to investigations of [Be75]. The region of unresolved resonances extends to 10 keV. The values of $\bar{\Gamma}_\gamma = 0.026$ eV, $D(l=0) = 40$ eV, $S_0 = 1.2 \cdot 10^{-4}$ were obtained from parameters of resolved resonances. The evaluation of p-wave strength function was performed by data from [Ly68]. Average fission width was evaluated so as to reproduce resonance integral evaluation from [Be72]. Average neutron width was evaluated by parameters of resolved resonances. Above 10 keV capture cross-section was calculated by statistical model according to data from [Ma76]. At the same energy range fission cross-section was calculated by the same model with normalizing by data from [Mo71]. The (n,2n) and (n,3n) reaction cross-sections were determined by the same model.

In Tables 7.43-7.49 characteristics of interaction between neutrons and ^{248}Cm obtained from various versions of evaluated data are compared with proposed and experimentally evaluated data.

Table 8.43

The values of σ_{f0} and $\sigma_{\gamma 0}$ for ^{248}Cm , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
σ_{f0}	0.341	0.370	0.087	$0.34^{+0.07}_{-0.07}$	1973, [5]
				$0.37^{+0.05}_{-0.05}$	1984, [6]
				$0.37^{+0.04*}_{-0.04*}$	1993, [3]
$\sigma_{\gamma 0}$	3.034	2.570	2.446	4^{+1}_{-1}	1973, [5]
				$2.63^{+0.26}_{-0.26}$	1984, [6]
				$2.59^{+0.26**}_{-0.26**}$	1974, [51]
				$2.63^{+0.26*}_{-0.26*}$	1973, [54]
				$3.4^{+1.0***}_{-1.0***}$	1993, [3]

*) - measurements (or measurement in case of $\sigma_{\gamma 0}$) on thermal columns

**) - differential evaluation

***) - measurement on thermal reactor spectra

Table 8.44

The values of I_f , I_γ for ^{248}Cm , in barn.

Value	ENDL-82	JENDL-3	ENDF/B-6	Recommend. value	Year of public.
I_f	14.24	15.69	13.63	$13.2^{+0.8}_{-0.8}$	1973, [5]
				15	1984, [6]
				$13.6^{+0.1*}_{-0.1*}$	1993, [3]
I_γ	283.8	260.4	246.7	275^{+75}_{-75}	1973, [5]
				270^{+15}_{-15}	1984, [6]
				259^{+12**}_{-12**}	1974, [51]
				$275.5^{+8.5*}_{-8.5*}$	1993, [3]
				275^{+75***}_{-75***}	1971, [45]

*) - activation measurement with Cd-shield

**) - differential evaluation, $E_{\max} = 3 \text{ keV}$

***) - reactor measurement

Table 8.45

The values of $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{248}Cm , in barn.

Evaluation	$\bar{\sigma}_f$	$\bar{\sigma}_\gamma$	$\bar{\sigma}_{2n}$	σ_f	σ_{2n}	σ_{3n}
ENDL-82	1.343	0.075	0.026	2.100	0.150	0.800
JENDL-3	1.257	0.052	0.012	2.680	0.148	0.236
ENDF/B-6	1.398	0.046	0.010	2.499	0.230	0.750

Table 8.46

The value of ΔI_f for ^{248}Cm , in barn.

$E_{\min}$ (eV)	$E_{\max}$	ENDL-82	JENDL-3	ENDF/B-6
0.5	10	4.980	7.409	4.070
10	10^2	3.559	3.326	4.253
10^2	10^3	0.743	0.324	0.205
10^3	10^4	0.191	0.136	0.060
10^4	10^5	0.188	0.105	0.090
10^5	10^6	0.677	0.588	0.580
10^6	10^7	3.907	3.805	4.371

Table 8.47

The value of ΔI_γ for ^{248}Cm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6
0.5 10	140.0	131.4	118.5
10 10^2	127.9	118.9	118.6
10^2 10^3	11.13	6.500	6.570
10^3 10^4	3.260	2.490	2.010
10^4 10^5	0.982	0.710	0.638
10^5 10^6	0.438	0.290	0.332
10^6 10^7	0.090	0.070	0.050

Table 8.48

The values of $\langle \sigma_f \rangle$ for ^{248}Cm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6
0.5 10	3.742	5.571	3.082
10 10^2	1.645	2.211	2.570
10^2 10^3	0.241	0.148	0.093
10^3 10^4	0.083	0.057	0.028
10^4 10^5	0.081	0.036	0.032
10^5 10^6	0.522	0.506	0.479
10^6 10^7	1.716	1.758	2.059

Table 8.49

The value of $\langle \sigma_{\gamma} \rangle$ for ^{248}Cm , in barn.

$E_{\min}$ $E_{\max}$ (eV)	ENDL-82	JENDL-3	ENDF/B-6
0.5 10	106.1	99.54	89.75
10 10^2	56.53	52.92	52.71
10^2 10^3	4.481	2.940	2.954
10^3 10^4	1.088	0.933	0.796
10^4 10^5	0.361	0.261	0.252
10^5 10^6	0.130	0.076	0.087
10^6 10^7	0.019	0.015	0.012

8.8 ^{249}Cm .

Even-odd ^{249}Cm is a short-lived nucleus ($T_{1/2} \approx 64.15$ min) transforming into ^{249}Bk due to β^- -decay. ^{249}Cm is produced at neutron capture by ^{248}Cm nucleus, thereat this reaction should go more intensively in flux of resonance neutrons with the energies from ≈ 0.5 eV to ≈ 100 eV. Because of small half-life value for ^{249}Cm the process of β^- decay should be the main transmutation channel of this nucleus under reactor conditions and processes of its interaction with neutrons may slightly compete with β^- -decay. Only one result of experimental evaluation of characteristics of interaction of neutrons with ^{249}Cm nucleus was published in literature. This is a result of $\sigma_{\gamma 0}$ measurement in thermal reactor flux, and it has been just this value suggested as proposed value in [5] and later conserved in [6].

Among the evaluated data versions compared by us the evaluations

of characteristics of interaction of neutrons with ^{249}Cm nucleus are given in JENDL-3 only. Some of these characteristics corresponding to this evaluation are presented in Tables 7.50 and 7.51. JENDL-3 (MAT 3969) evaluation was carried out by Y. Kikuchi and T. Nakagawa (JAERI, Japan). It was completed in 1984. At the energy range from 10^{-5} eV to 4.15 eV the $1/V$ law was adopted for cross-sections of all processes, that is, resonance cross-section structure is missing in evaluation. The region of unresolved resonances is between the limits 4.15 eV and 30 keV. The values of $S_0 = 1.08 \cdot 10^{-4}$, $S_1 = 3.95 \cdot 10^{-4}$, $S_2 = 1.04 \cdot 10^{-4}$, $R = 8.8$ Fermi were obtained by optical-model calculations. The values of $\bar{D} = 8.3$ eV and $\bar{\Gamma}_\gamma = 40$ MeV were evaluated from level density parameters. Average fission width was fitted so as to reproduce fission cross-section evaluation. The last was evaluated as $0.95 \cdot \sigma_f(^{247}\text{Cm})$ according to data from [Bi78]. At the energies above 30 keV capture cross-section was calculated by statistical model with the help of CASTHY code [Ig75] in terms of competitive processes and with the strength function of γ -quanta of $4.8 \cdot 10^{-4}$ evaluated by average resonance parameters in the region of unresolved resonances. The $(n,2n)$ and $(n,3n)$ reaction cross-sections were calculated by evaporation model in according with data from [Pe65].

In Tables 8.50-8.51 characteristics of interaction of neutrons with ^{249}Cm nucleus obtained from various versions of evaluated data are compared with recommended and experimentally evaluated data.

Table 8.50

The values of σ_{f0} , $\sigma_{\gamma0}$, I_f , I_γ , $\bar{\sigma}_f$, $\bar{\sigma}_\gamma$, $\bar{\sigma}_{2n}$, σ_f , σ_{2n} , σ_{3n} for ^{249}Cm , in barn.

Value	JENDL-3	Value	JENDL-3
σ_{f0}	0.820	$\bar{\sigma}_\gamma$	0.027
$\sigma_{\gamma0}^{*)}$	1.600	$\bar{\sigma}_{2n}$	0.036
I_f	137.4	σ_f	2.400
I_γ	215.1	σ_{2n}	0.190
$\bar{\sigma}_f$	2.220	σ_{3n}	0.481

*) - recommended value of 1.6 ± 0.8 barn

Table 8.51

The values of ΔI_f , ΔI_γ , $\langle \sigma_f \rangle$, $\langle \sigma_\gamma \rangle$ for ^{249}Cm , in barn.

JENDL-3 evaluation					
$E_{\min}$	$E_{\max}$	ΔI_f	ΔI_γ	$\langle \sigma_f \rangle$	$\langle \sigma_\gamma \rangle$
(eV)					
0.5	10	46.71	89.97	31.50	60.50
10	10^2	55.11	97.68	18.99	32.54
10^2	10^3	15.75	21.88	5.440	7.049
10^3	10^4	5.550	4.390	2.159	1.437
10^4	10^5	4.880	1.080	1.919	0.319
10^5	10^6	4.240	0.100	2.063	0.052
10^6	10^7	5.150	0.020	2.177	0.004

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^{242}Cm

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9.CONCLUSION.

The investigation of actinide nuclear property has been started more half century ago. Two prominent problems of twenty century - the creation of nuclear weapons and power reactors - were the principal stimulus for realization of such a long research program. The main attention was directed on acquisition of nuclear data for a small set

of nuclei used as power reactor fuel or nuclear weapon material or produced in noticeable amount by neutron capture. It is the main nuclear group, which includes the Th-232, U-233,235,238, Np-237 and Pu-238 - 242 nuclei. They have enough long half life and are produced in sufficient great amount. These circumstances alleviated their experimental investigation using particle accelerators and research reactors as neutron sources.

The second actinide nuclei group includes such long lived nuclei as Pa-231, U-232,234,236, Pu-244, Am-241,242m,243, Cm-243⁻ - 248, which are accumulated by neutron irradiation of nuclear fuel. They are less important for fuel cycle than first nuclei group and therefore they have not been explored adequately.

The third actinide nuclei group includes following short-lived nuclei: Pa-232,233, Th-233, U-237,239, Np-238,239, Pu-243, Am-242g,244g,244m and Cm-242,249, which are produced by transmutation of original fuel. The equilibrium amount of these nuclei in fuel is very small, therefore their neutron data has no influence practically on fuel cycle. Last time new nuclear power direction are discussed, for example, the problem of long lived radioactive waste (second group) along closed fuel cycle.

The transmutation facility differs from ordinary power reactors by more high level of neutron flux needed for high enough transmutation rate. The typical neutron flux is about $10^{15} - 10^{16}$ n/cm²·s for different estimations. In this case transmutation life time is about 1 - 10 days for capture cross-section of 10^3 barn. This life time is comparable with the half life for β -decay of the third nuclei group. Therefore it is necessary to take into account the competition of two branches: radioactive decay and nuclear reactions. The fuel

composition for transmutation facility also differs from power reactor fuel. Second group nuclei are the main fuel component for transmutation. The transmutation facility based on accelerator, target and subcritical blanket differs from ordinary power reactor also by original neutron spectrum formed by spallation (p,xn) reaction. Spallation neutrons are faster than fission spectrum ones (average neutron energy is = 20 MeV and = 2 MeV, respectively).

As a result for transmutation problem the main nuclear data requirements displace from the first to the second and third nuclei groups. The first nuclei group materials will be used as an amplifier of initial neutron flux, therefore it is necessary to know nuclear data particularly in fast neutron region.

We have considered evaluated nuclear data state for 37 actinide nuclei from the point of view of requirements arisen from transmutation problem. Evaluated neutron data from main modern libraries: ENDL-82, ENDF/B-6 (USA), JENDL-3 (JAPAN) and BROND-2 (former SU) and recommended data in ordinary energy region below 20 MeV were taken by us as a basis for analysis. We distinguished four specific subregion within it: thermal, resonance, fission spectrum and around 14 MeV. The region of thermal and resonance energy is more convenient and effective from the point of view of transmutation because of joint action of two factors: high level neutron flux (to 10^{16} n/cm².s) and high level values of capture and fission cross-section (to several thousands of barn). Fission process dominates in fission spectrum region. The threshold (n,xn) reaction give essential contribution (less than fission reaction) around 14 MeV region.

Neutron data for first group nuclei are the most completely and

thoroughly studied. They are included practically in all references libraries and handbooks. Maximal neutron cross-section differences are limited by 10%. This situation is rather successful. Neutron data for the second group nuclei are known with great discrepancies (up to 50%) and are not complete. These data should be refined and completed by direct measurements and nuclear model calculations. The most unsuccessful situation is for the third group nuclei. It is connected with a very difficult measurement conditions for short lived target nuclei and with a slight interest of these nuclear data. If the half life is much less than that of transmutation, nuclear data for short lived nuclei may be considered as not essential ("momentary" β -decay). It is necessary to take into account nuclei transmutation probability for high level neutron flux (to 10^{16} n/cm².s) and cross-section (to thousand barn) and half life more than 1 day. Unfortunately, neutron cross-section for short lived actinide nuclei are practically absent or measured near thermal point with great errors. The information about file presence in reference libraries is given in Table 1 for illustration. The values of $\sigma_i^{\max}/\sigma_i^{\min}$ ratio for different cross-sections showing maximal discrepancies between the values of different libraries are also included.

Ratio of fission cross-section at thermal point and resonance integrals of fission are particularly large in cases where there is fission threshold and fission probability is small. Capture cross-section ratio has large discrepancies in the region of fission spectrum, where capture cross-sections fast decrease with energy increasing. Average fission cross-section is determined well enough in this region. Maximal difference among average values are not more than 5% for the first nuclei group and 10% for the second one. Near 14 MeV

Table 9.1

Nuclear data information for actinides

nn	Nuclei	1	2	3	4	5	6	7	8	9	10	11	12
1.	Th-232	+	+	+	+	15,6	1,01	1,06	1,12	1,05	1,16	1,04	1,32
2.	Th-233	+	+	-	-	1	1,07	7,7	1,61	1,15	1,13	1,21	1,55
3.	Pa-231	-	+	+	-	3,3	1,14	76,5	2,86	1,18	1,38	1,06	1,83
4.	Pa-232	-	+	-	-	1	1,64	1	1,04	-	-	-	-
5.	Pa-233	-	+	+	-	-	1,06	2,24	1,05	1,45	1,55	1,66	2,79
6.	U-232	-	+	+	-	1,05	1,03	1,33	1,84	1,23	3,54	1,92	11,4
7.	U-233	+	+	+	+	1,01	1,05	1,06	1,06	1,05	1,19	1,04	2,50
8.	U-234	+	+	+	-	77,4	1,04	1,44	7,02	1,02	1,57	1,03	1,89
9.	U-235	+	+	+	+	1,03	1,05	1,03	1,14	1,01	1,05	1,02	1,56
10.	U-236	+	+	+	+	10,6	1,06	1,67	2,5	1,04	1,89	1,05	3,42
11.	U-237	+	-	+	-	5,7	1,25	10,6	4,14	1,07	1,29	1	1
12.	U-238	+	+	+	+	436	1,02	1,02	1,08	1,03	1,15	1,02	1,06
13.	U-239	+	-	-	-	1	1	-	-	-	-	-	-
14.	Np-237	+	+	+	-	1,43	1,07	1,15*	1,12	1,05	1,10	1,10	1,04
15.	Np-238	+	-	+	-	1,03	4,72	1,64	10,6	-	-	-	-
16.	Np-239	-	+	+	-	-	2,08	1	1,03	-	-	-	-
17.	Pu-238	+	+	+	+	1,10	1,02	1,38	1,15	1,04	1,45	1,03	4,70
18.	Pu-239	+	+	+	+	1,01	1,01	1,03	1,10	1,01	1,41	1,06	2,23
19.	Pu-240	+	+	+	+	2,13	1,05	1,17	1,06	1,08	1,18	1,07	4,6
20.	Pu-241	+	+	+	+	1,02	1,13	1,08	1,39	1,10	1,7	1,08	4,75
21.	Pu-242	+	+	+	+	2,84	1,04	9,1	1,17	1,11	1,21	1,10	1,69
22.	Pu-243	+	-	+	-	1,09	1,47	1,09	1,03	1	1	1	1
23.	Pu-244	-	-	+	-	-	1,08	-	2,6	-	-	-	-
24.	Am-241	+	+	+	+	1,09	1,58	1,81	1,25	1,08	2,3	1,18	2,9
25.	Am-242g	-	+	+	+	1,38	21,8	2,04	17,0	1,04	1,77	1,05	2,14
26.	Am-242m	+	+	+	+	1,10	1,59	1,34	1,76	1,21	5,95	1,07	2,14
27.	Am-243	+	+	+	+	2,7	1,09	1,67	1,31	1,05	4,46	1,05	11,0
28.	Am-244g	-	+	-	-	1	-	-	-	-	-	-	-
29.	Am-244m	-	+	-	-	1	-	-	-	-	-	-	-
30.	Cm-242	+	+	+	+	1,67	1,26	7,85	1,49	1,78	4,5	1,18	29,4
31.	Cm-243	+	+	+	-	1,15	6,75	2,53	2,07	1,07	6,1	1,16	6,45
32.	Cm-244	+	+	+	+	2,5	1,49	2,94	1,11	1,15	3,3	1,32	5,65
33.	Cm-242	+	+	+	-	1,22	1,16	1,16	1,21	1,06	1,56	1,18	1,44
34.	Cm-246	+	+	+	-	2,7	1,29	1,67	1,38	1,12	1,36	1,15	11,0
35.	Cm-247	+	+	+	-	1,30	1,58	1,53	2,2	1,08	1,76	1,02	2,41
36.	Cm-248	+	+	+	-	4,2	1,56	1,19	1,15	1,10	1,63	1,28	1,55
37.	Cm-249	-	+	-	-	-	-	-	-	-	-	-	-

Column names for table 9.1:

1) - ENDL-82; 2) - JENDL-3; 3) - ENDF/B-6; 4) - BROND-2;

5) - $\sigma_{f0}^{\max}/\sigma_{f0}^{\min}$; 6) - $\sigma_{\gamma 0}^{\max}/\sigma_{\gamma 0}^{\min}$; 7) - $I_f^{\max}/I_f^{\min}$; 8) - $I_\gamma^{\max}/I_\gamma^{\min}$;9) - $\bar{\sigma}_f^{\max}/\bar{\sigma}_f^{\min}$; 10) - $\bar{\sigma}_\gamma^{\max}/\bar{\sigma}_\gamma^{\min}$; 11) - $\sigma_f^{\max}/\sigma_f^{\min}$; 12) - $\sigma_{2n}^{\max}/\sigma_{2n}^{\min}$.

fission cross-section ratio has small difference (not more 10%). Situation in this region is much worse for the $(n,2n)$ and $(n,3n)$ cross-sections. Cross-section ratio of various libraries differ several times. Spallation neutron spectrum is very fast, therefore it is necessary to know the (n,xn) process probability with high precision in order to describe the conversion of original neutron spectrum to more slower one of secondary neutrons.

The transmutation rate for actinide nuclei is determined, first, by region of thermal and resonance neutrons, where neutron flux maximal level and fission and capture cross-sections may be reached. These data for the third nuclei group are unknown or determined roughly. Therefore it is necessary to carry out new experiments and investigations first and foremost for such nuclei as Pa-232,233, U-237, Np-238,239 and Am-242g in thermal and resonance region, where fission and capture cross-sections are high enough. There is good cross-section approximation by modern nuclear model calculations. Fast neutron contribution of fission and (n,xn) reaction to the common rate of actinide transmutation is small. Summary absorption cross-section is about 3 barn for 14 MeV that is several hundred times below than that in thermal and resonance regions. Fast neutron reactions are essential for creation of excess neutrons to good advantage for transmutation after moderation. Therefore it is necessary to know with high precision the cross-sections of neutron multiplication for corresponding materials (from the first or second groups), that will be used in target blanket systems.

Application of subcritical blanket with $K_{ef.}=0.95-0.98$ provides a possibility to increase neutron flux 20-50 times. However, in this case it is necessary to have easily accessible reliable subcritical

level for all stages of transmutation process with account of change of blanket composition. These conditions impose heavy demands on precision of neutron data needed for K_{ef} correct calculation. The choice of main blanket materials essentially depends on proposed conception. On the other hand, the choice of one or other blanket type is mainly determined by nuclear data for specific fissionable materials. In the light of transmutation problem the nuclear data status for actinides is considered to be unsatisfactory. Great work is to be done to create special-purpose library of evaluated nuclear data for actinides on a basis of new experimental data and calculations, that will permit to eliminate completely the main drawbacks and gaps in available data.

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