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NEUTRON EXCITATION FUNCTION GUIDE FOR REACTOR DOSIMETRY

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***Neutron Excitation Function Guide for Reactor Dosimetry** (NEFGRD)* has been prepared in the Ukrainian Nuclear Data Center (UKRNDC) using ZVV 9.2 code for graphical data presentation. The data can be retrieved through Web or obtained on CD-ROM or as hard copy report. NEFGRD contains graphical and text information for 56 nuclides (81 dosimetry reactions). Each reaction is provided by the information part and several graphical function blocks (from one to nine).

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NEFGRD contains graphical and text information for 56 nuclides (81 dosimetry reactions). Each reaction is provided by the information part and several graphical function blocks (from one to nine). Graphs were obtained as PS-files using ZVV9.2 code and then were converted to PDF-files. The list of reactions is contained in the ***Table 1*** below. Adobe Acrobat Reader is necessary to display files in PDF format.

Web access to the data. The reaction field of ***Table 3*** (**ContTable**) of Web version of this report has hypertext links to the graphical and textual components of each of the reaction included. The hypertext links are indicated by blue color. The report with data is available from the IAEA Nuclear Data Section Web-site:
http://www-nds.iaea.org (as in January 2002).

Data on CD-ROM. NEFGRD is available on CD-ROM as one file in the PDF format and links from reaction field to the beginning of the data presentation for particular reaction in the report.

Data in the form of hardcopy report. The report is available as hardcopy print out on the request sent to the IAEA Nuclear Data Section (see NDS addresses at the cover pages).

Total size of the report is about 28 Mbytes. The size of the file for each reaction is shown in ***Table 1***.

Table 1. The list of the reactions with links to data and size of files.

Reaction	Size (b)	Reaction	Size (b)	Reaction	Size(b)
Ag109ngm	79 552	Hg199nnṁ	32 610	S32np	81 330
Al27na	159 000	I127n2n	53 831	Sc45ng	367 507
Al27np	160 698	In115n2n	74 082	Ta181ng	1 006 227
Am241nf	905 504	In115ngm12	531 604	Th232nf	292 487
As75n2n	51 530	In115nnṁ	96 590	Th232ng	783 099
Au197n2n	62 507	La139ng	347 416	Ti46n2n	72 040
Au197ng	934 652	Li6na	156 438	Ti46np	119 760
B10na	91 000	Li6nt	214 706	Ti47nnp	68 274
Cdng	473 754	Li7nt	88 643	Ti47np	119 399
Co59n2n	81 468	Mg24np	129 128	Ti48nnp	68 814
Co59na	117 193	Mn55n2n	63 072	Ti48np	120 977
Co59ng	395 326	Mg55ng	377 970	Ti49nnp	64 601
Cr50ng	575 934	Na23n2n	56 372	TinxSc46	44 886
Cr52n2n	72 288	Na23ng	320 522	TinxSc47	45 268
Cu63n2n	72 852	Nb93n2n	89 501	TinxSc48	44 092
Cu63na	71 988	NB93ng	977 456	Tm169n2n	53 420
Cu63ng	652 810	Nb93nnṁ	42 298	U235nf	6 082 152
Eu151ng	464 684	Ni58n2n	125 133	U238nf	1 873 000
F19n2n	90 613	Ni58np	159 532	U238ng	1 917 658
Fe54n2n	71 099	Ni60np	79 575	V51na	87 991
Fe54na	98 142	Np237nf	1 042 112	W186ng	596 556
Fe54np	130 151	O16n2n	63 646	Y89 n2n	61 102
Fe56np	141 655	P31np	90 209	Zn64np	117 697
Fe57nnp	63 113	Pb204nnṁ	28 361	Zr90n2n	96 716
Fe58na	64 954	Pr141n2n	54 719	B10aprod	96 704
Fe58ng	371 828	Pu239nf	4 009 144	C12 n2n	48 015
Gd0ng	205 800	Rh103nnṁ	47 778	Cu65n2n	82 695
				SUM	**27.8 MB**

Graphical information. The graphical information consists of dosimetry reaction cross sections as the function of neutron energy taken from available data of specialized dosimetry libraries IRDF-90 [1], D-99 [2], RRDF-98 [3], of general purpose evaluated nuclear data libraries ENDF/B-VI [4], JENDL-3.2 [5], JEF-2.2 [6], BROND-2 [7], CENDL-2 [8] and experimental cross sections data in EXFOR format from Cross Section Information Storage and Retrieval System .

Point-wise reaction cross section data from ENDF libraries were taken using the following 3 sources:

1) Internet-site http://www-nds.iaea.org/endf/
(marked by * in **ContTable**);
2) Internet-site http://www-nds.iaea.org/zvd/
(marked by ** in **ContTable**);
3) "ENDF PACKAGE, October 1999", CD ROM produced by the IAEA
(marked by + in **ContTable**).

Point-wise cross sections from the last source were calculated using the NJOY94.61 nuclear data processing system [9] with fractional error tolerances of 0.1%.

The most (n,γ) and (n,f) reactions with complex resonance structure are presented on the graphs as group averaged (not point-wise) cross sections, except for the first (general) graph. The beginning and the end of the averaged area were defined individually for each reaction, but in all cases the group boundaries are the same as in the SAND- II (640 group energy structure). The GROUPIE program from the PREPRO code package [10] was used to calculate these averaged cross sections. In the case of such averaging, in the graph after the name of the corresponding ENDF-library the letter G was placed (in brackets). All cross sections taken from definite dosimetry or evaluated library are shown in all graphs with the color assigned to this library:

- IRDF-90 - solid green
- D99 - solid light blue
- RRDF 98 - solid violet
- ENDF/B-VI - solid light red
- JENDL-3.2 - dotted light violet
- JEF-2.2 - dotted red
- BROND-2 - long-short dotted blue
- CENDL-2 - dotted light green

The only exceptions are 3-Li-6 and 5-B-10 alpha production reactions where several reactions from the same library are given in different colors in the graphs.

Experimental data for most reactions were taken from Internet-site: http://www-nds.iaea.org/exfor/. For 21 reactions listed in the Table 2 below, the experimental data were taken from CD-ROM "EXFOR II/ACCESS-97 (Ver.06B-Test, March 2000)".

Table 2. Experimental data retrieved from EXFOR-II/ACCESS-97 CD-ROM (Version 0.6B-Test, March 2000)

Reaction	Reaction	Reaction
3-Li-6 (n, t) 2-He-4	22-Ti-00 (n, x) 21-Sc-48	69-Tm-169 (n, 2n) 69-Tm-168
3-Li-6 Alpha production	22-Ti-49 (n, np) 21-Sc-48	79-Au-197 (n, γ) 79-Au-198
3-Li-7 Tritium production	23-V-51 (n, α) 21-Sc-48	80-Hg-199 (n, n') 80-Hg-199m
5-B-10 Alpha production	26-Fe-58 (n, α) 24-Cr-55	82-Pb-204 (n, n') 82-Pb-204m
8-O-16 (n, 2n) 8-O-15	40-Zr-90 (n, 2n) 40-Zr-89	90-Th-232 (n, f)
22-Ti-00 (n, x) 21-Sc-46	57-La-139 (n, γ) 57-La-140	92-U-235 (n, f)
22-Ti-00 (n, x) 21-Sc-47	59-Pr-141 (n, 2n) 59-Pr-140	95-Am-241 (n, f)

Experimental data at the graphs are presented by various blue symbols. Mostly, experimental data uncertainties are also shown, if they exist. In the cases of very large amount of data, the uncertainties were omitted and the experimental points were given as light blue symbols.

All graphs are supplied with short legend with references ordered by year of publication and name of first author. More detailed information is given in **Tabl. 3** of the information part (see below).

For alpha production reactions the experimental data in graphs are presented by symbols of various colors for different reactions. In these cases the additional information is presented at the graphs.

Additionally, there are shown by vertical lines the energy boundaries corresponding to 10, 50 and 90% response function for U-235 thermal neutron spectrum, calculated using ENDF/B-VI data (if they are). Alternatively, there were used the other libraries: JENDL-3.2, JEF-2, BROND-2, CENDL-2, IRDF-90, D-99, RRDF-98 (the order of libraries corresponds to the priority). The detailed information on response function see below.

From one to nine graphs are given for each reaction depending on complexity of cross section structure.

Information block. The information block for each reaction consists of four parts.

The first block contains the following information:

- The name of reaction.
- The abundance and/or the half-life of the target nuclide.
- The reaction Q-values and the threshold energy.
- The half-life of the reaction product.
- The energy and the absolute intensity of gamma-rays or/and absolute intensity of betas and maximum energy of the beta-spectrum and decay mode.

The reaction Q-values and the energy of threshold are calculated using internet-site http://t2.lanl.gov/data/qtool.html. Abundance and decay modes are taken from [11]. Half-lives, energies and intensities of gamma-rays or/and intensity of betas and maximum energy of the beta-spectrum are taken from [12, 13]. One or several gamma-lines with the largest intensities are chosen. The uncertainty of half-live is given as last significant figures given after units. For example, 35.60 h 6 means $T_{1/2}$=35.60 ±0.06 h.

The second information block gives name of library, date and authors of the last evaluation of available libraries for the given reaction.

The third block consists of two tables - **Tabl. 1** and **Tabl. 2**. Each table gives the calculated energy boundaries of the 10, 50, 90% response function and average cross sections for all available libraries. All energy boundaries are given in MeV, average cross sections are given in barn. For these calculations two neutron spectra are used: the U-235 thermal neutron fission spectrum and Cf-252 spontaneous fission spectrum. Both spectra are taken from the IRDF-90 library (for Cf-252 MT=1, U-235 for MT=2). These calculations were done with own code CR_SP.

The fourth block (**Tabl. 3**) includes six column information on experimental data, as follows:

column 1 energy range
column 2 number of experimental points
column 3 number of the regional nuclear data centre, country and institute (laboratory)
column 4 publication type (R-report, T-thesis, J-journal, C- conference, etc. All

abbreviations are explained e.g. in CINDA)

column 5	year and month (or the year only) and the author or the first co-author (with comma)
column 6	accession number under which the entry is stored in the EXFOR format library.

Table 3. **ContTable**, Table with (hyper) links to the data

The ContTable is given at the next page.

№	Reaction	MT	Dosimetry File			Evaluated File				
			IRDF-90	D-99	RRDF-98	ENDF/B-VI	JENDL-3	JEF-2	BROND-2	CENDL-2
1.	3-Li-6 (n, t) 2-He-4	105	+	+		*	*	*	*	
2.	3-LI-6 Alpha production	207		+						
3.	3-Li-7 Tritium production	205		+			+	+		+
4.	5-B-10 (n, α) 3-Li-7	107	+	+		*	*	*		*
5.	5-B-10 Alpha production	207		+						
6.	6-C-12 (n, 2n) 6-C-11	16			+					
7.	8-O-16 (n, 2n) 8-O-15	16			+		*			*
8.	9-F-19 (n, 2n) 9-F-18	16	+	+	+	*	*	*	*	*
9.	11-Na-23 (n, 2n) 11-Na-22	16		+		*	*	*	*	*
10.	11-Na-23 (n, γ) 11-Na-24	102	+	+		+	+	+	+	+
11.	12-Mg-24 (n, p) 11-Na-24	103	+	+	+	**	**			
12.	13-Al-27 (n, p) 12-Mg-27	103	+	+		*	*	*		*
13.	13-Al-27 (n, α) 11-Na-24	107	+	+		**	**	*		**
14.	15-P-31 (n, p) 14-Si-31	103	+	+		*	*	*	*	*
15.	16-S-32 (n, p) 15-P-32	103	+	+		*	*	*		
16.	21-Sc-45 (n, γ) 21-Sc-46	102	+	+		+	+			
17.	22-Ti (n, x) 21-Sc-46	220		+						
18.	22-Ti (n, x) 21-Sc-47	221		+						
19.	22-Ti (n, x) 21-Sc-48	222		+						
20.	22-Ti-46 (n, 2n) 22-Ti-45	16		+	+		**			
21.	22-Ti-46 (n, p) 21-Sc-46	103	+	+	+	**	**			
22.	22-Ti-47 (n, np) 21-Sc-46	28	+	+	+	*	*			
23.	22-Ti-47 (n, p) 21-Sc-47	103	+	+		*	*			
24.	22-Ti-48 (n, np) 21-Sc-47	28	+	+	+	*	*			
25.	22-Ti-48 (n, p) 21-Sc-48	103	+	+	+	*	*			
26.	22-Ti-49 (n, np) 21-Sc-48	28		+	+		+			
27.	23-V-51 (n, α) 21-Sc-48	107	+(00)		+(51)	*(00)		*(00)		*(00)
28.	24-Cr-50 (n, γ) 24-Cr-51	102		+		+	+	+	+	+
29.	24-Cr-52 (n, 2n) 24-Cr-51	16	+	+		*	*	*	*	
30.	25-Mn-55 (n, 2n) 25-Mn-54	16	+	+		*	*	*		*
31.	25-Mn-55 (n, γ) 25-Mn-56	102	+	+		*	+	+		+
32.	26-Fe-54 (n, 2n) 26-Fe-53	16			+	*	*	*	*	*
33.	26-Fe-54 (n, p) 25-Mn-54	103	+	+		*	*	*	*	+
34.	26-Fe-54 (n, α) 24-Cr-51	107			+	*	*	*	*	+
35.	26-Fe-56 (n, p) 25-Mn-56	103	+	+	+	*	*	*	*	
36.	26-Fe-57 (n, np) 25-Mn-56	28		+		**	**	**		**
37.	26-Fe-58 (n, γ) 26-Fe-59	102	+	+		*	+	+	+	+
38.	26-Fe-58 (n, α) 24-Cr-55	107				**	**	**	**	**
39.	27-Co-59 (n, 2n) 27-Co-58	16	+	+		**	**	**		**
40.	27-Co-59 (n, γ) 27-Co-60	102	+	+		**	+	+		+
41.	27-Co-59 (n, α) 25-Mn-56	107	+	+	+	*	*	*		*

№	Reaction	MT	Dosimetry File			Evaluated File				
			IRDF-90	D-99	RRDF-98	ENDF/B-VI	JENDL-3	JEF-2	BROND-2	CENDL-2
42.	28-Ni-58 (n, 2n) 28-Ni-57	16	+	+		*	*	*	*	
43.	28-Ni-58 (n, p) 27-Co-58	103	+	+		*	*	*	*	
44.	28-Ni-60 (n, p) 27-Co-60	103	+	+		**	**	**	**	
45.	29-Cu-63 (n, 2n) 29-Cu-62	16	+	+		*	*			+
46.	29-Cu-63 (n, γ) 29-Cu-64	102	+	+		+	+			+
47.	29-Cu-63 (n, α) 27-Co-60	107	+	+	+	*	*			+
48.	29-Cu-65 (n, 2n) 29-Cu-64	16	+	+		*	*			+
49.	30-Zn-64 (n, p) 29-Cu-64	103	+	+				*		
50.	33-As-75 (n, 2n) 33-As-74	16			+		*			
51.	39-Y-89 (n, 2n) 39-Y-88	16	+	+		*	*	*		
52.	40-Zr-90 (n, 2n) 40-Zr-89	16	+	+		*	*		*	
53.	41-Nb-93 (n, 2n) 41-Nb-92m	16	+	+	+					
54.	41-Nb-93 (n, n') 41-Nb-93m	51	+	+	+					
55.	41-Nb-93 (n, γ) 41-Nb-94	102	+			*	+	+	+	+
56.	45-Rh-103 (n, n') 45-Rh-103m	51	+	+	+					
57.	47-Ag-109 (n, γ) 47-Ag-110m	102	+	+						
58.	48-Cd (n, γ)	102	+			+	+	+		+
59.	49-In-115 (n, 2n) 49-In-114m	16	+				*			
60.	49-In-115 (n, n') 49-In-115m	51	+	+	+	*				
61.	49-In-115 (n, γ) 49-In-116(m_1+m_2)	102	+	+		+	+	+		
62.	53-I-127 (n, 2n) 53-I-126	16	+	+		*	*	*		
63.	57-La-139 (n, γ) 57-La-140	102				+	+	+		
64.	59-Pr-141 (n, 2n) 59-Pr-140	16			+	**	**	**		
65.	63-Eu-151 (n, γ) 63-Eu-152	102		+		**	**	**		
66.	64-Gd (n, γ)	102	+						+	
67.	69-Tm-169 (n, 2n) 69-Tm-168	16		+						
68.	73-Ta-181 (n, γ) 73-Ta-182	102		+		+	+	+	+	+
69.	74-W-186 (n, γ) 74-W-187	102		+		+	+	+	+	
70.	79-Au-197 (n, 2n) 79-Au-196	16	+	+		*		*	*	*
71.	79-Au-197 (n, γ) 79-Au-198	102	+	+		+		+	+	+
72.	80-Hg-199 (n, n') 80-Hg-199m	51		+						
73.	82-Pb-204 (n, n') 82-Pb-204m	51								
74.	90-Th-232 (n, f)	18	+	+		*	*	+	*	*
75.	90-Th-232 (n,γ) 90-Th-233	102	+	+		+	+	+	+	
76.	92-U-235 (n, f)	18	+	+		+	+	+	+	
77.	92-U-238 (n, f)	18	+	+		+	+	+	+	+
78.	92-U-238 (n, γ) 92-U-239	102	+	+		+	+	+	+	+
79.	93-Np-237 (n, f)	18	+	+		+	+	+		+
80.	94-Pu-239 (n, f)	18	+	+		+	+	+	+	+
81.	95-Am-241 (n, f)	18		+		*	+	+	+	+

Comments:

+ –data from:

Dosimetry file:

IRDF-90 –data from internet site http://iaeand.iaea.or.at/ndspub/libraries/irdf/

D-99 – data from "JENDL Data & Figures" CD-rom (JENDL Dosimetry Fil e 99 (JENDL/D-99))

RRDF-98 – data from internet site http://rndc.ippe.obninsk.ru/data/rrdf/rrdf.htm

Evaluated File – "ENDF PACKAGE October 1999" CD-rom (ENDF Libraries, Processing codes, Manuals and Utilities)

* –data from internet site http://www-nds.iaea.or.at/endf/endfframe.html

** –data from internet site http://iaeand.iaea.or.at/zvd/

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