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HIGHLIGHTED REQUESTS OF NEUTRON NUCLEAR DATA MEASUREMENTS

For Fission Reactors

January 1977

Compiled by

WRENDA Working Group of Japanese Nuclear Data Committee

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Highlighted Requests of Neutron Nuclear Data Measurements for Fission Reactors

Compiled by

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Highlighted requests were selected from lists of requests for neutron nuclear Data measurements. This work was made by WRENDA Working Group of Japanese Nuclear Data Committee in response to an action of the NEANDC. This will be submitted to a NEA meeting on request lists. Compiled requests in this note correspond finally to the parts of the WRENDA 76/77 with some modifications and additional comments.

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中性子核データの測定に対する特に強い要望のリスト
核分裂炉用

シグマ研究委員会WRENDAグループ*) 編

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中性子核データの測定に対する要望の中から特に強く要望されているものを選び出してNEAの「測定に対する要求リストに関する委員会」へ提出した。この作業はNEANDCの要請に応えてシグマ研究委員会のWRENDAワーキンググループが行ったもので，最終的にはWRENDA76／77に登録されている要望の中から選んだものに若干の修正とコメントを加えた内容になっている。

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Japanese Nuclear Data Committee (JNDC) has continued to contribute to the WRENDA activity since 1968. A Working Group for WRENDA activity for fission reactors has been convened every year to screen new requests and to examine the old Japanese requests which had been compiled in WRENDA library. Requests for safeguards and fusion have been processed by other Working Groups of the JNDC. In 1975, the Working Group for fission reactors received about 100 new requests and reduced them by one-half through screening. The old requests registered in WRENDA 75 were examined whether they should be left, modified or deleted. After this, the Working Group compiled a Japanese list of requests in 1975 which is composed of 163 requests including the remaining old requests and adopted new ones, and presened it to WRENDA 76/77. In the present note, we picked up from these requests what were considered to be most urgently needed for fission reactors. The following is a brief description of the reason behind requests presented in this note.

A Working Group on fission Product Nuclear Data in JNDC had stressed the need of the measurements for the neutron capture cross section of the following thirteen nuclides in the energy region from 100 eV to 400 keV : Tc-99, Ru-101, Pd-105, Pd-107, Ag-109, Xe-131, Cs-133, Cs-135, Nd-143, Nd-145, Pm-147, Sm-149 and Sm-151. The reasons for the choice of these nuclides are the following:

(1) they are of primary importance for prediction of the effect of the fission products on the fast reactors,

(2) no experimental data of capture cross sections exist except in the resonance energy region,

and

(3) there are large discrepancies among existing data as in the case of Ag-109 and Cs-133.

A Working Group on Compilation of Japanese Evaluated Nuclear Data Library has discussed various problems about the status of the experimental data and presented the follwoing requests. The fission cross section of U-235 plays a central role on the evaluation of heavy element nuclear data which were obtained by the relative measurements to the U-235 fission cross section. Though the discrepancies among recent data are fairly reduced, more accuracy than 2% is still required for determination of the energy dependence, in particular, in the region 100 keV to 1 MeV.

The total cross section gives apparently the upper limit of the partial cross sections and is reproduced by the theoretical calculations rather easily. Inversely, the experimental total cross section is used to determine the model parameters in the theoretical calculations which are also useful for estimation of some partial cross sections. In this sense, the total cross sections of Pu-240 and Pu-241 are required in the present note. Besides, they may be useful for estimation of the unresolved resonance parameters. The neutron capture cross section of Pu-240 is needed for safety problems of the fast reactors after high burn up. Status of the data is unsatisfactory for these problems as well as for irradiation tests of the fuel.

Build-up of transplutonium nuclides in spent fuel is an important problem in the transportation and reprocessing of the spent fuel. In particular, estimation of the build-up of Am-242 is necessary for neutron shielding design of the transport cask of the spent fuel, because of the neutron emission through spontaneous fission. For this estimation, the neutron capture cross section of Am-241 must be known, but the present status of the data is very poor. Therefore, it is emphatically wanted to measure the capture cross section of Am-241. Besides, the capture cross sections of Am-243, Cm-242, Cm-243 and Cm-244 are necessary for estimation of the build-up of heavier nuclides. The fission cross section of Am-241 must be measured in order to reduce the large discrepancy below 500 keV.

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NO.	NUCLIDE	QUANTITY	ENERGY(EV) MIN	ENERGY(EV) MAX	ACCURACY (%)
1	^{99}Tc	N,GAMMA	1.2+2	4.0+5	20.0
	Comment : For fast reactor calculations. Data by Chou et al. do not agree with integral data.				
2	^{101}Ru	N,GAMMA	1.0+2	4.0+5	20.0
	Comment : For fast reactor calculations. No experimental data above 100 eV.				
3	^{105}Pd	N,GAMMA	1.0+2	4.0+5	20.0
	Comment : For fast reactor calculations. Discrepancies exist among experimental data.				
4	^{107}Pd	N,GAMMA	1.0+2	4.0+5	20.0
	Comment : For fast reactor calculations. No experimental data above 100 eV.				
5	^{109}Ag	N,GAMMA	1.0+2	4.0+5	30.0
	Comment : For fast reactor calculations. Large discrepancies exist among experimental data.				
6	^{131}Xe	N,GAMMA	1.0+2	4.0+5	30.0
	Comment : For fast reactor calculations. No experimental data above 100 eV.				
7	^{133}Cs	N,GAMMA	1.0+2	4.0+5	20.0
	Comment : For fast reactor calculations. The experimental data are abundant in the energy range below 15 MeV, but systematic discrepancies are observed in the range above 10 keV. Recent data by Yamamuro et al. indicate low values at 24 keV. Absolute measurement is further required.				
8	^{135}Cs	N,GAMMA	1.0+2	4.0+5	20.0
	Comment : For fast reactor calculations. No experimental data above 100 eV.				
9	^{143}Nd	N,GAMMA	1.0+2	4.0+5	30.0
	Comment : For fast reactor calculations. No experimental data above 100 eV. Large discrepancies among evaluated data.				

NO.	NUCLIDE	QUANTITY	ENERGY(EV) MIN	MAX	ACCURACY (%)
10	^{145}Nd	N,GAMMA	1.0+2	4.0+5	30.0
	Comment :	For fast reactor calculations. No experimental data above 100 eV.			
11	^{147}Pm	N,GAMMA	1.0+2	4.0+5	30.0
	Comment :	For fast reactor calculations. No experimental data above 100 eV.			
12	^{149}Sm	N,GAMMA	1.0+2	4.0+5	20.0
	Comment :	For fast reactor calculations. No experimental data except a measurement at 30 keV.			
13	^{151}Sm	N,GAMMA	1.0+2	4.0+5	30.0
	Comment :	For fast reactor calculations. No experimental data above 100 eV.			
14	^{233}Pa	N,GAMMA	2.0+1	1.5+7	10.0
	Comment :	For burn-up calculation of Th fueled thermal reactors. No experimental data.			
15	^{235}U	FISSION	1.0+3	2.0+7	2.0
	Comment :	For evaluation of the nuclear data on U-235, and design calculation for thermal and fast reactors. Absolute measurement wanted. Discrepancies between the experimental data are very remarkable in the energy range below 70 keV, because of strong energy dependence in this region. Measurements with monochromatic neutron beam desired.			
16	^{239}Pu	FISSION	1.0+3	1.5+7	2.0
	Comment :	For fast reactor calculations. Discrepancies still exist among recent experimental data.			
17	^{240}Pu	TOTAL	6.0+3	1.5+7	5.0
	Comment :	For fast reactor calculations. No experimental data between 6 and 100 keV, and above 1.5 MeV. Data are required for performing self-consistent evaluation.			
18	^{240}Pu	N,GAMMA	1.0+3	2.0+6	10.0
	Comment :	For fast reactor calculations. Insufficient experimental data. No measurement above 350 keV.			

NO.	NUCLIDE	QUANTITY	ENERGY(EV) MIN	MAX	ACCURACY (%)
19	^{241}Pu	TOTAL	2.0+3	1.5+7	5.0

Comment : For fast reactor calculations. No experimental data above 2 keV. Data are required for performing self-consistent evaluation.

NO.	NUCLIDE	QUANTITY	ENERGY(EV) MIN	MAX	ACCURACY (%)
20	^{241}Pu	ALPHA	1.0-1	2.0+6	8.0

Comment : For fast reactor calculations. Insufficient experimental data.

NO.	NUCLIDE	QUANTITY	ENERGY(EV) MIN	MAX	ACCURACY (%)
21	^{242}Pu	N,GAMMA	1.0+3	2.0+6	10.0

Comment : For shielding of spent fuel. Insufficient experimental data. No experimental data above 100 keV.

NO.	NUCLIDE	QUANTITY	ENERGY(EV) MIN	MAX	ACCURACY (%)
22	^{241}Am	N,GAMMA	1.0+2	2.0+6	10.0

Comment : For estimation of build-up of transplutonium nuclides in spent fuel. Production cross sections of the ground and isomer states of Am-242. Neutron shielding design for transport cask of spent fuel. Insufficient experimental data.

NO.	NUCLIDE	QUANTITY	ENERGY(EV) MIN	MAX	ACCURACY (%)
23	^{241}Am	FISSION	1.0+2	1.0+7	10.0

Comment : For estimation of build-up of transplutonium nuclides in spent fuel. Neutron shielding design for transport cask of spent fuel. Large discrepancies exist among experimental data below 500 keV. Insufficient data above 5 MeV.

NO.	NUCLIDE	QUANTITY	ENERGY(EV) MIN	MAX	ACCURACY (%)
24	^{242}Am	N,GAMMA	THR	2.0+6	30.0

Comment : For estimation of build-up of transplutonium nuclides in spent fuel. (n,γ) cross sections of the ground and isomer states of Am-242. Neutron shielding design for transport cask of spent fuel. Insufficient experimental data.

NO.	NUCLIDE	QUANTITY	ENERGY(EV) MIN	MAX	ACCURACY (%)
25	^{243}Am	N,GAMMA	2.0+1	2.0+6	20.0

Comment : For estimation of build-up of transplutonium nuclides in spent fuel. Production cross sections of the ground and isomer states of Am-244. Neutron shielding design for transport cask of spent fuel. Insufficient experimental data.

NO.	NUCLIDE	QUANTITY	ENERGY(EV) MIN	MAX	ACCURACY (%)
26	^{242}Cm	N,GAMMA	THR	2.0+6	30.0
	Comment :	For estimation of build-up of transplutonium nuclides in spent fuel. Neutron shielding design for transport cask of spent fuel. No experimental data.			
27	^{242}Cm	FISSION	THR	1.5+7	30.0
	Comment :	For estimation of build-up of transplutonium nuclides in spent fuel. Neutron shielding design for transport cask of spent fuel. Insufficient experimental data.			
28	^{243}Cm	N,GAMMA	1.0+0	2.0+6	30.0
	Comment :	For estimation of build-up of transplutonium nuclides in spent fuel. Neutron shielding design for transport cask of spent fuel. No experimental data.			
29	^{244}Cm	N,GAMMA	5.0+1	2.0+6	10.0
	Comment :	For estimation of build-up of transplutonium nuclides in spent fuel. Neutron shielding design for transport cask of spent fuel. No experimental data.			