GROUP CONSTANTS FOR ^{233}U, ^{235}U and ^{239}Pu IN THE RESONANCE REGION

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At present we are seeing a slowing down in the arrival of new experimental information on nuclear data for reactor materials. This is said to be due to the difficulty of setting up essentially new experiments, and also to the partial meeting of reactor nuclear data requirements. In this situation there is an urgent need to develop calculation methods and programs for the preparation of group constants based on neutron cross-section parameter information. The present paper gives the results of group constant calculations for three fissile nuclides prepared using the method of joint analysis of neutron cross-sections and transmission functions in a multilevel model. Reference [1] describes the methods for analysing the neutron cross-sections of heavy nuclides in the unresolved resonance region, and Ref. [2] does the same for the resolved region [2]. The latter analysis led to an evaluation of resonance parameters for ^{239}Pu [3].

The analytical approach was generally as follows - the parameters of the theoretical model were evaluated on the basis of experimental information on mean neutron cross-sections and transmission functions; then, group constants were calculated using an accurate theoretical model. The parameter sensitivity coefficients for all functionals (transmission and self-indication functions, group constants) were calculated simultaneously. The new features of the approach were as follows:

- Careful statistical evaluation (mean values and covariance matrices) of mean resonance parameters and group constants;
- Use of a suitably accurate theoretical model which allows for inter-level interference effects;
- Incorporation in the analysis of experimental information on transmission and self-indication functions, which makes for significantly greater accuracy in the resonance self-shielding factors.

In this way, self-consistent evaluations of the values and errors of the resonance parameters and group constants were obtained. In the nuclear data evaluation literature for fissile nuclides in the unresolved resonance region, the authors stay within the framework of a simplified theoretical model of the Hauser-Feshbach type which does not allow resonance self-shielding factors to be evaluated and does not take inter-level interference into account. Therefore, in existing group constant tables (see, for example, Ref. [4]) the evaluations of mean cross-sections, on the one hand, and of resonance self-shielding factors and their temperature dependence, on the other, are not self-consistent.

The present paper gives evaluations of group constants in the unresolved resonance region for ^{233}U, ^{235}U, and ^{239}Pu drawing on up-to-date recommended data on mean cross-sections and transmission functions (the latter were measured for ^{235}U and ^{239}Pu). For ^{239}Pu, these group constant evaluations extend right up to the thermal neutron energy region. This was made possible thanks to the use of an improved S-matrix formalism when evaluating the resonance parameters. Experimental data for other fissile nuclides in the resolved region are not good enough to permit an analogous approach to be employed with the same degree of reliability.

Available data on mean resonance parameters, and in particular, for ^{233}U, the evaluations given in Ref. [5], were used as a priori information in the unresolved resonance region. Note that there is no information in Ref. [4] on this nuclide which is so important for research into the thorium cycle. For ^{235}U, we worked from the mean cross-section data on which the evaluations in ENDF/B-V [6] are based, and for ^{239}Pu, from the data given in Ref. [4], since they are the most up to date.

Mean resonance parameters. The data in Ref. [5] were used for ^{233}U, and no fitting was done owing to the lack of experimental information on mean cross-sections. These parameters do not depend on energy: the mean radiation width $\bar{\Gamma}_\gamma$, which is the same in all states and equals 0.039 eV; the

potential scattering radius R' = 9.93 fm; and the mean distance between S-resonances $\bar{D}$ = 6.80 eV. The law of proportionality was assumed for the different states J : $D_J \approx 1(2J + 1)$. The remaining parameters (the strength functions S_0, S_1 and the fission widths) are given in the form of fluctuating values in a fine-group partition for the 0.1-30 keV range.

The mean resonance parameters for ^{235}U were optimized in the light of the experimental data on the transmission and self-indication functions for the fission reaction given in Ref. [7]. The following evaluations were obtained for the parameters which do not depend on energy: the mean radiation width $\bar{\Gamma}_\gamma = 30 \pm 2$ MeV (for all states), the p-strength function $S_1 = 1.68 \pm 0.45 \times 10^{-4}$, and the mean distance between S-resonances $\bar{D} = 440 \pm 20$ MeV. Including the transmission functions in the calculations meant that the monotonic energy dependence of the potential scattering radius R' on the neutron energy had also to be introduced. In addition, it was found that the observed mean cross-sections could only be described using fluctuating values (from group to group) for the S-strength function and fission widths. The average value for the S-strength function over a wide lethargy interval was found to be $S_0 = 0.97 \pm 0.05 \times 10^{-4}$, and the scattering radius $\bar{R}' = 9.3 \pm 0.1$ fm.

An analogous optimization process was performed for ^{239}Pu. The experimental data used here are given in Ref. [8], where preliminary evaluations are also given of the parameters and of certain constants for ^{239}Pu (done by "manual" fitting). The present paper gives definitive statistical evaluations of: $\Gamma_\gamma = 39.5 \pm 4.0$ MeV, $S_1 = 2.17 \pm 0.40 \times 10^{-4}$ (these parameters are not dependent on neutron energy), $\bar{R}' = 9.2 \pm 0.2$ fm (this parameter is monotonically dependent on neutron energy), $\bar{S}_0 = 0.98 \pm 0.06 \times 10^{-4}$ (fluctuating parameter). The fission widths varied from group to group.

The fluctuation in the strength functions indicate that the processes which occur when neutrons interact with fissile nuclei are highly complex. Experience shows, that for even-even (non-fissile) nuclei, a good description

Table 1

Mean-group cross-sections for the nuclides ^{239}Pu, ^{235}U, ^{233}U in the unresolved resonance region, burns

Group No.	E, keV	^{239}Pu				^{235}U				^{233}U			
		σ_t	σ_f	σ_c	σ_{el}	σ_t	σ_f	σ_c	σ_{el}	σ_t	σ_f	σ_c	σ_{el}
11	10-21,5	14,77	1,82	0,932	12,02	14,70	2,60	1,10	11,0	16,40	3,14	0,562	12,7
12	4,65-10	16,51	2,14	1,573	12,8	16,39	3,42	1,37	11,6	18,04	4,20	0,739	13,1
13	2,15-4,65	19,15	2,92	2,53	13,7	18,82	5,12	1,70	12,0	22,37	7,02	1,35	14,0
14	1-2,15	22,9	4,35	4,07	14,5	22,44	7,13	2,91	12,4	27,36	10,8	2,16	14,4
15	0,465-1	29,97	8,14	6,23	15,6	28,48	11,4	4,58	12,5	30,64	13,4	2,94	14,3
16	0,215-0,465	-	-	-	-	36,30	16,2	7,40	12,7	35,04	17,6	3,34	14,1
17	0,100-0,215	-	-	-	-	46,6	21,6	11,5	13,5	58,49	34,8	8,79	14,9

of the mean cross-sections can be obtained with constant strength function values.

Group constants for ^{233}U, ^{235}U and ^{239}Pu in the unresolved resonance region. Table 1 gives the mean-group cross-sections for the nuclides under investigation (in the ABBN groups [4]). They differ but little from the primary data on which the mean neutron cross-section evaluations are based. Our results differ from the data given in Ref. [6] for ^{235}U by no more than 5%, and from those given in Ref. [4] for ^{239}Pu by no more than 10%.

The following tables give information which is based on a new methodology (resonance self-shielding factors as a function of dilution cross-section and temperature). These data differ from those in Ref. [4] in that they have been obtained using a unified procedure for evaluating the primary information (mean cross-sections and transmission functions) and a suitably accurate theoretical model.

Table 2 gives data on resonance self-shielding factors for three nuclides for different dilution cross-sections at room temperature. When these data are compared with the results given in Ref. [4] it is seen that, for ^{235}U, they are systematically lower, and that for ^{239}Pu the opposite obtains. The temperature (Doppler) increments for these factors are given in Table 3. They are significantly lower than the corresponding data in Ref. [4]. The difference is particularly large (factor of 1.5-2) for a zero

Table 2

Resonance self-shielding factors as a function of dilution cross-section at a temperature of 300 K (in burns)

Group No.	$f_t(\sigma_0)$				$f_f(\sigma_0)$				$f_c(\sigma_0)$				$f_{el}(\sigma_0)$			
	0	10	10^2	10^3	0	10	10^2	10^3	0	10	10^2	10^3	0	10	10^2	10^3
							^{233}U									
II	0,897	0,935	0,984	0,994	0,931	0,957	0,990	0,999	0,926	0,953	0,989	0,999	0,951	0,970	0,993	0,999
I2	0,888	0,927	0,981	0,993	0,910	0,941	0,985	0,998	0,898	0,932	0,982	0,998	0,954	0,971	0,993	0,999
I3	0,818	0,870	0,960	0,992	0,831	0,880	0,965	0,996	0,803	0,858	0,957	0,995	0,944	0,962	0,989	0,999
I4	0,776	0,826	0,934	0,967	0,784	0,836	0,944	0,992	0,736	0,796	0,926	0,990	0,952	0,964	0,987	0,998
I5	0,712	0,766	0,900	0,963	0,714	0,775	0,914	0,987	0,661	0,726	0,887	0,982	0,956	0,966	0,986	0,998
I6	0,667	0,723	0,871	0,960	0,688	0,749	0,896	0,984	0,591	0,657	0,841	0,972	0,963	0,971	0,987	0,998
I7	0,461	0,516	0,702	0,911	0,544	0,604	0,787	0,953	0,441	0,498	0,696	0,924	0,903	0,915	0,948	0,985
							^{235}U									
II	0,889	0,933	0,984	0,999	0,927	0,957	0,990	0,999	0,924	0,955	0,990	0,999	0,948	0,969	0,993	0,999
I2	0,877	0,924	0,981	0,998	0,909	0,943	0,987	0,999	0,905	0,941	0,986	0,999	0,948	0,968	0,993	0,999
I3	0,855	0,905	0,974	0,997	0,874	0,917	0,978	0,998	0,864	0,909	0,975	0,997	0,954	0,971	0,993	0,999
I4	0,809	0,861	0,955	0,991	0,819	0,871	0,961	0,995	0,799	0,855	0,955	0,995	0,956	0,969	0,991	0,999
I5	0,697	0,764	0,909	0,962	0,729	0,795	0,928	0,990	0,703	0,772	0,917	0,988	0,944	0,958	0,985	0,998
I6	0,573	0,645	0,832	0,952	0,622	0,697	0,873	0,980	0,599	0,673	0,858	0,976	0,727	0,941	0,973	0,995
I7	0,478	0,543	0,747	0,931	0,548	0,622	0,822	0,967	0,484	0,557	0,770	0,952	0,922	0,933	0,964	0,992
							^{239}Pu									
II	0,843	0,902	0,975	0,998	0,895	0,936	0,985	0,998	0,861	0,914	0,979	0,998	0,922	0,952	0,989	0,999
I2	0,779	0,852	0,956	0,998	0,822	0,855	0,970	0,997	0,776	0,852	0,960	0,995	0,894	0,930	0,981	0,998
I3	0,713	0,788	0,922	0,993	0,734	0,813	0,943	0,993	0,654	0,749	0,918	0,989	0,871	0,907	0,968	0,996
I4	0,634	0,698	0,859	0,986	0,618	0,706	0,889	0,984	0,516	0,617	0,847	0,977	0,849	0,880	0,949	0,992
I5	0,498	0,555	0,732	0,906	0,482	0,569	0,794	0,961	0,354	0,445	0,714	0,943	0,789	0,818	0,898	0,977

dilution cross-section. Hence, we may conclude that the positive component of the Doppler coefficient of reactivity for fissile nuclei is in reality significantly smaller than it appears when calculated from the constants given in Ref. [4], which is a matter of some importance for reactor construction.

Group constants for ^{239}Pu in the resolved resonance region.

Reference [2] describes a method for the joint evaluation method of neutron cross-sections for the nuclide ^{239}Pu in the resolved resonance region using an improved S-matrix theory. This evaluation procedure is special in that the results are tested against measurements of the averaged fission reaction transmission and self-indication functions (as with the results above for the unresolved resonance region). This makes for greater reliability when calculating resonance self-shielding factors and their temperature dependence.

Table 3

Doppler increments of resonance self-shielding factors for ^{233}U as a function of dilution cross-section

Group No.	$\Delta f_t(\sigma_0)$				$\Delta f_f(\sigma_0)$				$\Delta f_c(\sigma_0)$				$\Delta f_{e\ell}(\sigma_0)$			
	0	10	10^2	10^3	0	10	10^2	10^3	0	10	10^2	10^3	0	10	10^2	10^3
							^{233}U									
11	0,007	0,004	0,001	0,001	0,010	0,007	0,001	0	0,012	0,008	0,002	0	0,002	0,001	0	0
	0,004	0,002	0,001	0	0,006	0,004	0,001	0	0,006	0,004	0,001	0	0,001	0	0	0
12	0,014	0,010	0,003	0,001	0,021	0,014	0,004	0,001	0,025	0,017	0,005	0,001	0,003	0,002	0,001	0
	0,008	0,005	0,001	0	0,011	0,007	0,002	0,002	0,013	0,006	0,002	0	0,001	0,001	0	0
13	0,026	0,021	0,008	0,002	0,035	0,026	0,009	0,001	0,046	0,035	0,012	0,002	0,005	0,004	0,002	0
	0,017	0,013	0,005	0,001	0,022	0,016	0,005	0,001	0,026	0,019	0,006	0,001	0,003	0,002	0,001	0
14	0,041	0,035	0,017	0,003	0,050	0,039	0,015	0,002	0,055	0,024	0,004	0,001	0,008	0,006	0,003	0,001
	0,033	0,027	0,011	0,002	0,036	0,028	0,010	0,001	0,037	0,014	0,002	0,001	0,006	0,005	0,002	0
15	0,045	0,041	0,024	0,005	0,056	0,047	0,022	0,004	0,075	0,065	0,033	0,060	0,007	0,006	0,003	0
	0,039	0,034	0,017	0,003	0,045	0,036	0,015	0,002	0,056	0,047	0,021	0,034	0,005	0,004	0,002	0
16	0,033	0,033	0,025	0,006	0,045	0,040	0,022	0,004	0,068	0,064	0,041	0,009	0,004	0,003	0,002	0,001
	0,033	0,031	0,019	0,004	0,041	0,035	0,017	0,003	0,056	0,050	0,027	0,005	0,003	0,003	0,002	0
17	0,035	0,038	0,040	0,019	0,051	0,048	0,035	0,011	0,067	0,068	0,012	0,022	0,006	0,006	0,006	0,003
	0,041	0,042	0,036	0,013	0,051	0,047	0,030	0,008	0,064	0,061	0,044	0,013	0,006	0,006	0,005	0,002
							^{235}U									
11	0,007	0,004	0,001	0,001	0,009	0,005	0,001	0	0,009	0,006	0,001	0	0,002	0,001	0	0
	0,003	0,002	0,001	0	0,004	0,003	0,001	0	0,005	0,003	0,001	0	0,001	0	0	0
12	0,012	0,008	0,002	0,002	0,016	0,010	0,003	0	0,017	0,011	0,003	0	0,003	0,002	0	0
	0,007	0,004	0,001	0,001	0,009	0,006	0,001	0	0,010	0,006	0,002	0	0,001	0,001	0	0
13	0,022	0,016	0,005	0,002	0,030	0,021	0,006	0,001	0,031	0,022	0,007	0,001	0,004	0,003	0,001	0
	0,014	0,010	0,003	0,002	0,018	0,012	0,003	0	0,019	0,013	0,004	0	0,003	0,002	0,001	0
14	0,039	0,031	0,012	0,003	0,048	0,036	0,012	0,002	0,053	0,045	0,042	0,002	0,006	0,004	0,002	0
	0,026	0,020	0,007	0,002	0,030	0,022	0,007	0,001	0,033	0,020	0,019	0,001	0,004	0,003	0,001	0
15	0,060	0,052	0,026	0,006	0,070	0,056	0,023	0,034	0,078	0,064	0,027	0,004	0,009	0,008	0,004	0,001
	0,043	0,035	0,015	0,003	0,044	0,034	0,012	0,017	0,047	0,036	0,014	0,002	0,006	0,005	0,002	0
16	0,072	0,068	0,044	0,011	0,087	0,073	0,036	0,007	0,097	0,084	0,044	0,008	0,010	0,009	0,006	0,002
	0,061	0,053	0,029	0,006	0,063	0,052	0,023	0,004	0,066	0,055	0,025	0,004	0,008	0,007	0,004	0,001
17	0,062	0,066	0,058	0,019	0,088	0,080	0,048	0,011	0,097	0,091	0,061	0,016	0,010	0,009	0,007	0,003
	0,069	0,065	0,043	0,011	0,076	0,064	0,032	0,006	0,083	0,073	0,042	0,009	0,009	0,008	0,006	0,002
							^{239}Pu									
11	0,020	0,014	0,004	0,002	0,020	0,013	0,003	0,001	0,030	0,020	0,006	0	0,010	0,007	0,002	0
	0,011	0,008	0,002	0,001	0,012	0,007	0,002	0	0,015	0,010	0,002	0,001	0,005	0,004	0,001	0
12	0,033	0,025	0,010	0,004	0,034	0,023	0,007	0	0,054	0,039	0,012	0,002	0,017	0,013	0,005	0
	0,020	0,015	0,005	0,002	0,021	0,014	0,004	0,001	0,029	0,020	0,006	0,001	0,010	0,007	0,002	0,001
13	0,045	0,041	0,023	0,007	0,055	0,043	0,016	0,002	0,090	0,074	0,030	0,004	0,024	0,021	0,010	0,001
	0,031	0,027	0,012	0,004	0,038	0,029	0,010	0,001	0,061	0,038	0,013	0,002	0,015	0,012	0,005	0,001
14	0,048	0,050	0,037	0,011	0,069	0,060	0,031	0,005	0,107	0,096	0,050	0,008	0,025	0,022	0,013	0,002
	0,042	0,039	0,023	0,007	0,054	0,044	0,018	0,003	0,074	0,062	0,027	0,005	0,010	0,017	0,009	0,002
15	0,037	0,044	0,054	0,023	0,067	0,068	0,049	0,012	0,094	0,098	0,077	0,021	0,020	0,021	0,018	0,006
	0,041	0,045	0,042	0,014	0,066	0,060	0,036	0,007	0,084	0,081	0,051	0,011	0,019	0,018	0,014	0,003

Note: Numerator - Δ_1 = f (300 K), denominator - Δ_2 = f (2100 K) - f (900 K).

Table 4 gives the calculated results for ^{239}Pu mean-group cross-sections, and Table 5, the resonance self-shielding factor results as a function of temperature and dilution cross-section for the same nuclide. These calculations were made using the GRUKON program [9] which has a high

Table 4

Mean cross-sections for ^{239}Pu in the resolved resonance region, burns

Group No.	E_n, eV	σ_t	σ_f	σ_c	σ_{el}	Group No.	E_n, eV	σ_t	σ_f	σ_c	σ_{el}
16	215-465	41,9	13,1	13,3	15,5	22	2,15-4,65	24,4	11,5	4,8	8,1
17	100-215	52,5	19,6	17,5	15,4	23	1,00-2,15	35,5	23,4	3,6	8,6
18	46,5-100	115,6	57,0	40,4	18,2	24	0,465-1,00	149,6	99,5	40,3	9,8
19	21,5-46,5	69,0	22,8	34,4	11,8	25	0,215-0,465	2762,3	1699,0	1052,5	10,8
20	10,0-21,5	189,7	104,6	73,5	11,5	Thermal	0,0253	1019,6	744,0	269,1	6,5
21	4,65-10,0	68,6	33,8	27,0	7,8						

level of accuracy thanks to the careful selection of the quadrature formulae for finite-difference integration. The following results were obtained. Group cross-sections on the whole agreed with the data given in Ref. [4], except for the data on $\langle\sigma_f\rangle$ and $\langle\sigma_c\rangle$ in groups 18 and 19. Note that the values obtained for $\langle\sigma_t\rangle$ were systematically higher those given in Ref. [4]. At the same time, the mean cross-sections obtained showed a high level of agreement with reliable primary experimental data on $\sigma_t(E)$ [10] and $\sigma_f(E)$ [11] used in the optimization procedure. The close agreement of the data obtained for $\langle\sigma_\gamma\rangle$ with the measurement results familiar from Ref. [12] should also be noted, though the latter were not taken into account in the evaluation process. Finally, the resonance self-shielding factors $f_t(\sigma_0)$ and $f_f(\sigma_0)$, at room temperature, proved to be very close to the values obtained by direct processing of the measurement results for the fission reaction transmission and self-indication functions [13]. At the same time, these results differ from those given in Ref. [4] in the same way as those for the unresolved resonance region (the evaluations obtained for $f_f(\sigma_0)$ and $f_c(\sigma_0)$ being systematically higher than the data in Ref. [4]).

When comparing the results obtained on temperature changes in the resonance self-shielding factors with those given in Ref. [4], differences analogous to those for the unresolved resonance region are evident. For the fission reaction, the calculated temperature increments for the intervals 900-300 K and 2100-900 K are systematically and significantly smaller than the

Table 5

Resonance self-shielding factors for ^{239}Pu, burns

Group No.	E_n, eV	T, K	$f_t(\sigma_0)$					$f_f(\sigma_0)$					$f_c(\sigma_0)$					$f_{el}(\sigma_0)$				
			0	I0	$I0^2$	$I0^3$	$I0^4$	0	I0	$I0^2$	$I0^3$	$I0^4$	0	I0	$I0^2$	$I0^3$	$I0^4$	0	I0	$I0^2$	$I0^3$	$I0^4$
I6	2I5–465	300	0,486	0,522	0,669	0,902	0,987	0,570	0,623	0,788	0,95I	0,994	0,35I	0,4I2	0,640	0,9II	0,989	0,8I0	0,827	0,888	0,970	0,996
		900	0,525	0,569	0,735	0,937	0,992	0,628	0,682	0,839	0,968	0,996	0,446	0,5I3	0,736	0,946	0,994	0,835	0,852	0,9I4	0,98I	0,998
		2I00	0,573	0,623	0,793	0,958	0,996	0,686	0,737	0,877	0,978	0,998	0,542	0,608	0,809	0,965	0,996	0,860	0,877	0,935	0,988	0,999
I7	I00–2I5	300	0,38I	0,4I9	0,555	0,820	0,962	0,458	0,5II	0,689	0,908	0,987	0,293	0,346	0,548	0,850	0,979	0,744	0,759	0,820	0,93I	0,989
		900	0,399	0,443	0,604	0,866	0,980	0,493	0,55I	0,737	0,934	0,992	0,350	0,4II	0,629	0,897	0,987	0,758	0,776	0,844	0,950	0,993
		2I00	0,426	0,476	0,656	0,90I	0,984	0,539	0,599	0,783	0,932	0,994	0,4I5	0,48I	0,698	0,928	0,99I	0,775	0,795	0,868	0,964	0,996
I8	46,5–I00	300	0,2I3	0,24I	0,365	0,654	0,927	0,328	0,375	0,552	0,824	0,970	0,I99	0,230	0,369	0,70I	0,946	0,579	0,597	0,666	0,83I	0,968
		900	0,2I8	0,249	0,387	0,709	0,949	0,346	0,395	0,58I	0,855	0,978	0,223	0,26I	0,425	0,770	0,964	0,593	0,6I3	0,693	0,868	0,979
		2I00	0,23I	0,266	0,423	0,762	0,963	0,376	0,429	0,6I9	0,883	0,984	0,262	0,306	0,492	0,825	0,975	0,6I6	0,637	0,725	0,899	0,985
I9	2I,5–46,5	300	0,I69	0,I87	0,264	0,544	0,903	0,I46	0,I85	0,340	0,702	0,950	0,087	0,I20	0,267	0,65I	0,959	0,725	0,740	0,786	0,895	0,98I
		900	0,I70	0,I90	0,280	0,6I2	0,93I	0,I55	0,I98	0,376	0,755	0,963	0,098	0,I38	0,3I3	0,722	0,958	0,728	0,745	0,800	0,9I8	0,987
		2I00	0,I73	0,I96	0,307	0,683	0,95I	0,I69	0,220	0,425	0,806	0,973	0,II5	0,I64	0,372	0,785	0,970	0,734	0,753	0,8I7	0,937	0,99I
20	I0,0–2I,5	300	0,096	0,II4	0,202	0,507	0,888	0,I47	0,I86	0,346	0,690	0,943	0,I05	0,I36	0,278	0,643	0,934	0,803	0,806	0,83I	0,9I2	0,983
		900	0,095	0,II4	0,206	0,553	0,9I3	0,I5I	0,I92	0,365	0,729	0,955	0,II0	0,I45	0,305	0,697	0,950	0,804	0,808	0,837	0,926	0,988
		2I00	0,095	0,II4	0,2I7	0,607	0,933	0,I58	0,203	0,393	0,768	0,965	0,I2I	0,I6I	0,346	0,750	0,963	0,806	0,8II	0,847	0,940	0,99I
2I	4,65–I0,0	300	0,270	0,279	0,323	0,533	0,887	0,247	0,266	0,363	0,663	0,935	0,I94	0,2I4	0,3I8	0,639	0,930	0,947	0,947	0,953	0,974	0,995
		900	0,270	0,279	0,325	0,567	0,909	0,249	0,269	0,377	0,70I	0,949	0,I97	0,2I8	0,333	0,679	0,945	0,947	0,948	0,954	0,977	0,996
		2I00	0,27I	0,279	0,330	0,6II	0,928	0,254	0,276	0,399	0,743	0,960	0,20I	0,225	0,356	0,725	0,957	0,947	0,948	0,955	0,980	0,997
22	2,I5–4,65	300	0,987	0,99I	0,998	I,00	I,00	0,985	0,990	0,997	I,00	I,00	I,00	I,00	I,00	I,00	I,00	0,999	0,999	I,00	I,00	I,00
		900	0,987	0,99I	0,998	I,00	0,999	0,985	0,990	0,997	I,00	I,00	I,00	I,00	I,00	I,00	I,00	0,999	0,999	I,00	I,00	I,00
		2I00	0,987	0,99I	0,998	I,00	I,00	0,985	0,990	0,997	I,00	I,00	I,00	I,00	I,00	I,00	I,00	0,999	0,999	I,00	I,00	I,00
23	I,00–2,I5	300	0,932	0,945	0,980	0,997	0,999	0,953	0,962	0,987	0,998	I,00	0,952	0,96I	0,986	0,998	I,00	0,996	0,997	0,999	I,00	I,00
		900	0,93I	0,945	0,980	0,997	0,999	0,953	0,962	0,987	0,998	I,00	0,952	0,96I	0,986	0,998	I,00	0,996	0,997	0,999	I,00	I,00
		2I00	0,93I	0,944	0,980	0,997	0,999	0,952	0,962	0,987	0,998	I,00	0,952	0,96I	0,986	0,998	I,00	0,996	0,997	0,999	I,00	I,00
24	0,465–I,00	300	0,556	0,575	0,684	0,905	0,988	0,720	0,736	0,820	0,953	0,994	0,6II	0,633	0,748	0,933	0,992	0,966	0,968	0,979	0,995	0,999
		900	0,553	0,57I	0,680	0,903	0,987	0,7I8	0,734	0,8I8	0,952	0,994	0,608	0,630	0,745	0,932	0,992	0,966	0,968	0,979	0,995	0,999
		2I00	0,546	0,564	0,673	0,899	0,987	0,7I2	0,728	0,8I3	0,950	0,994	0,60I	0,623	0,739	0,929	0,99I	0,965	0,967	0,978	0,994	0,999
25	0,2I5–0,465	300	0,333	0,336	0,360	0,522	0,854	0,588	0,59I	0,6I3	0,735	0,926	0,575	0,577	0,600	0,727	0,923	I,05	I,05	I,05	I,02	I,00
		900	0,342	0,345	0,369	0,534	0,863	0,60I	0,603	0,625	0,747	0,93I	0,588	0,59I	0,6I3	0,738	0,928	I,05	I,05	I,05	I,03	I,00
		2I00	0,36I	0,364	0,390	0,559	0,878	0,626	0,628	0,650	0,767	0,939	0,6I4	0,6I7	0,639	0,760	0,937	I,06	I,06	I,05	I,03	I,0I

values given in Ref. [4]. For the capture reaction, analogous differences may be observed in energy groups 18 and 21; in the remaining groups the difference is not so great. This shows that calculating the Doppler effect for ^{239}Pu on the basis of the data given in Ref. [4] leads to significant overestimation of that effect.

These results demonstrate the potential of the new programs developed by the Power Physics Institute for analysing neutron data and preparing group constants using evaluated resonance parameters within a multilevel formalism. These programs are of practical importance, particularly for the self-consistent calculation of group constants for fissile nuclei (in an arbitrary group representation) in the resolved and the unresolved resonance regions. The group constant evaluations obtained fill a gap in the data given in Ref. [4] for ^{233}U, and for ^{235}U and ^{239}Pu they show how data must be adjusted, particularly data on resonance self-shielding factors and their temperature dependence.

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