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COMPENDIUM OF SELECTED TRANSLATIONS OF SOVIET REPORTS ON THE EVALUATION OF THE NEUTRON NUCLEAR DATA FOR ^{235}U

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EVALUATION OF NUCLEAR CONSTANTS FOR ^{235}U IN THE NEUTRON ENERGY REGION BETWEEN 10^{-4} eV AND 15 MeV

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ABSTRACT

The authors briefly describe the results of an evaluation of a complete nuclear constant file for ^{235}U in the neutron energy region between 10^{-4} eV and 15 MeV. For evaluation purposes all available experimental data for ^{235}U were analysed in detail, and a number of methods of evaluating constants were developed and applied. The evaluated data for ^{235}U have been presented in the SOKRATOR format, and sent to the Obninsk Nuclear Data Centre.

^{235}U, one of the most important nuclei for reactor engineering, is at the same time one of the most difficult to analyse, both because the resonance levels are very close to each other, and because there is a large quantity of often quite contradictory experimental information.

In order to compile a complete file of the nuclear constants for ^{235}U, we analysed all available experimental information on this nucleus, developed a number of cross-section evaluation methods, and made the necessary calculations. The results of this evaluation were described in detail in a comprehensive report, submitted to the Nuclear Data Centre Bulletin, where it will be published in shortened form, and in a number of articles in the Nuclear Data Centre Bulletin. The present paper is only a short exposition of the evaluation methods used. The evaluated data for ^{235}U themselves have been presented in the SOKRATOR format and sent to the Nuclear Date Centre.

In the thermal region of neutron energies from 10^{-4} eV to 1 eV, we analysed the following quantities, which are of the main interest for the development of thermal neutron reactors:

σ_t, σ_α, σ_f, α and η. The method we applied to evaluate the thermal constants for ^{235}U was the same as the one used for ^{239}Pu. In particular, we compared the values of $\eta(E)$ measured directly, and $\eta(E)$ obtained from the ratio of σ_f to σ_α. All available experimental data in the 10^{-4}–1 eV region were analysed for four values of σ_t, from which σ_α is obtained by subtracting the elastic scattering cross-section, σ_f, η and α in the 10^{-4} – 1.0 eV energy region. Comparison of the derived and directly measured $\eta(E)$-curves shows that the derived curve agrees with the experimental data on η with an accuracy of the order of $\pm$ 3%, which is apparently the actual accuracy of measuring η.

In evaluating the resonance parameters of ^{235}U the idea at first was to use the resonance parameter values published by the experimenters or the authors of other evaluations. However, the experimenter as a rule analyses only the results of his own experiments. Available evaluations are individual in character and apply to a limited energy region. Besides, the results of evaluations are often contradictory, so one has to analyse the body of experimental results itself, try to expose its defects and obtain the resonance parameters anew. Such an approach was used in the present work. The available body of experimental data on σ_t, σ_f and σ_γ (^{235}U) was analysed critically, flaws in experimental technique were discovered in some work, and the selected data were normalized uniformly and corrected, where necessary, for the energy shift of the scale.

The selected sets of experimental data on σ_t, σ_f and σ_γ were analysed by means of a modified Adler-Adler formalism in order to obtain self-consistent Adler parameters and multi-level Breit-Wigner parameters.

The mean resonance parameters of ^{235}U in the 0.1 - 100 keV region were obtained by analysing the resonance parameters in the resolved resonance energy region and by fitting the calculated values of the cross-sections σ_t and σ_f to the experimental data in the unresolved resonance energy region. With mean parameters of ^{235}U one can calculate all types of cross-section. In order to check the quality of the mean parameters, we compared the calculated and experimental data on α.

In evaluating σ_f (^{235}U) we analysed the available experimental data including recently published data. We used the method of simultaneous evaluation of σ_f (^{239}Pu), the $\sigma_f(^{239}Pu)/\sigma_f$ (^{235}U) ratio and σ_f (^{235}U), and also analysed the correlated experimental errors.

Legendre polynomial expansion is the commonly accepted method of analysing the angular distributions of elastically scattered neutrons because of its mathematical simplicity and also because Legendre polynomials appear in the Schrödinger equation in the treatment of the scattering problem. At a sufficiently high level of Legendre polynomial expansion, however, the fitted curve can behave quite unphysically in the spaces between the experimental points, giving negative cross-section values. Moreover, scattering experiments in principle do not permit measurement of differential cross-sections at extremely small and extremely large angles, and therefore the fitting procedure must allow extrapolation to these angles. Owing to the orthogonality of the Legendre polynomials this is impossible, although scattering at small angles constitutes a large portion of the entire cross-section at energies exceeding 8 MeV. For a description of the angular distributions we therefore used Bessel function expansion, which requires significantly fewer terms than Legendre polynomial expansion. Moreover, no previous knowledge is needed of the differential scattering cross-sections at angles of 0° and 180°, the selection of which is often based on common-sense considerations. We used values of the scattering cross-sections at these angles obtained by Bessel function fitting to obtain the Legendre polynomial expansion. We used Bessel function expansion only for the intermediate stage, as description of angular distributions by Legendre polynomials is the commonly accepted procedure.

Experimental information on the inelastic neutron scattering cross-section is slight and therefore the evaluation had to be based on theoretical calculations, which were also made difficult by the competition from fission. We developed a formalism and wrote a programme that permits calculation of $\sigma_{nn'}$ (^{235}U) taking into account the competition from fission and capture.

Data on the spectra of γ-rays from capture are entirely lacking; for this reason, the spectra of such γ-rays were calculated on the basis of statistical theory in the present work. We wrote a programme corresponding to a statistical model in terms of which the spectrum of γ-rays from neutron capture were calculated. The contribution of direct and semi-direct capture processes becomes considerable at incident neutron energies exceeding 7 MeV, but at these energies the main contribution to the total γ-quantum spectrum comes from γ-rays from inelastic neutron scattering, as σ_{nn} is much larger than σ_{γ} in this region. Experimental information on the spectra of γ-rays from inelastic scattering is entirely lacking, so our evaluation was based solely on calculations. Unlike those from capture, γ-rays from inelastic scattering are difficult to calculate because the initial excitation function is not known, and has to be found first. We evaluated this excitation when evaluating the inelastic scattering. The total quantity of γ-quanta, their mean energy and the total energy removed by them was calculated.

It is practically impossible to calculate the spectra of γ-rays from fission without experimental information because the excitation density R (E) in the fragments is completely unknown. Besides, fission does not involve two fragments of constant mass - it has its own mass distribution (its level densities), a situation which makes the calculations more revealing, more useful for explaining tendencies. Fortunately, experimental work is available on γ-rays from fission: the measurements by Verbinkij et al. [1], Peele et al. [2], Rau et al. [3], Pleasonton et al. [4]. Analysis of these papers showed that Ref. [1], on which our evaluation is based, can be of real value for evaluation. Apart from the partial γ-ray spectra, the cross-sections of γ-quantum production and their average energy as a function of incident neutron energy were also calculated.

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RESONANCE PARAMETERS FOR ^{235}U IN THE ENERGY REGION UP TO 140 eV

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ABSTRACT

The authors analyse the available experimental data on σ_T, σ_f and σ_γ for ^{235}U in the energy region up to 140 eV. They obtain multi-level self-consistent resonance parameters and Breit-Wigner parameters permitting simultaneous calculation of all three types of cross-section. They give the average statistical parameters $\langle D\rangle$, $\langle\Gamma_\gamma\rangle$, $\langle\Gamma\rangle$ and $\langle\Gamma_f\rangle$.

As a rule, the available evaluations of the resonance parameters for ^{235}U are individual in character, and refer to a limited energy region. Moreover, the results of the evaluations are often contradicting, and the evaluations do not use all available experimental data. For this reason we have analysed all experimental information on the resonance energy region. As the ^{235}U nucleus is very complicated to analyse ($\langle D \rangle \sim 0.6$ eV), we used an Adler-Adler-type formalism for parametrization.

In order to parametrize the cross-sections σ_T, σ_f and σ_γ from 1 to 140 eV we investigated the following groups of experimental data:

I. on σ_T =

(1) Michaudon et al. [1] - data with good resolution at a temperature of 77° K, used throughout the region;

(2) Shore and Sailor [2] - measurements up to 10 eV, sufficiently reliable data, used in our paper;

(3) Brooks et al. [3] - the data do not describe the shape of the resonances satisfactorily and there is a significant shift in the energy scale; not used in our paper;

(4) Gerasimov et al. [4] - measurements up to 2.2 eV, not used;

(5) Simpson et al. [5] - insufficient resolution, no correction for ^{238}U impurities, data not used;

(6) Uttley [6] - no detailed information on energy resolution, data not used;

(7) Rainwater et al. [7] - no detailed information on resolution, irregular shift of energy scale, data do not describe shape of resonances satisfactorily, not used in our paper.

II. on σ_f:

(1) de Saussure et al. [8] - resolution not good enough, data not used;

(2) Blons [9] - measurements at a temperature of 77° K with good resolution, renormalization coefficient 1.0034; data in 40-140 eV range used, as it is difficult to evaluate the background reliably below 35 eV in experiment;

(3) Cao et al. [10] - data with good resolution up to 40 eV, renormalization coefficient 1.058, used for our paper in spite of low accuracy in inter-resonance region due to molybdenum screen;

(4) Deruytter et al. [11] - measurements up to 10.9 eV at sample temperature of 298°K with good resolution, data are absolute and the most accurate, used without renormalization;

(5) Shore and Sailor [2] - measurements up to 10 eV, data used with renormalization coefficient 0.974;

(6) Lemley et al. [12] - explosion data without detailed information on resolution; a number of fluctuations that are difficult to explain are found in the data, so they were not used;

(7) Brooks et al. [3] - see above;

(8) Michaudon et al. [1] - experimental resolution was poorer than in Blons's work [9], data not used; all experimental data were normalized to the integrals of Deruytter et al. [11].

III. on σ_γ:

(1) de Saussure et al. [8] - data used in 1-140 eV range in spite of insufficiently good resolution, renormalization coefficient 0.954 obtained by evaluation of α in 0.1 - 1 keV region;

(2) Brooks et al. [3] - see above;

(3) Perez et al. [13] - measurements from 8 eV, resolution of the same order as that of deSaussure et al. [8], data used with renormalization coefficient 1.01.

IV. on σ_n:

(1) Sauter et al. [14] - measurements in the 1 - 31 eV region, no information on sample temperature and resolution, the authors themselves consider the data unsuitable for multi-level analysis;

(2) Ceulemans and Poortmans [15] - no information on sample temperature, even in the 20 eV area the energy interval does not enable one to identify the levels with confidence.

The data on σ_n were not used for our paper.

To obtain the resonance parameters we used a modified Adler-Adler formalism, retaining its formal notation:

$$\sigma_{nr}(E) = \frac{2.6 \cdot 10^6}{E} \sum_{i=1}^{N} \left[G_i^{(r)} \psi(x,\theta) + H_i^{(r)} \chi(x,\theta) \right],$$

where N is the number of resonances taken into account, $G_i^{(r)}$ and $H_i^{(r)}$ are the Adler parameters of the i-level of the r-reaction, $\psi(x,\theta)$ and $\chi(x,\theta)$ are the Doppler functions, the following relation holding for $G_i^{(r)}$: $G_i^{(r)} = \dfrac{g_i \Gamma_{ni} \Gamma_{ri}}{\Gamma_i^2}$

where g_i, Γ_{ni}, Γ_{ri} and Γ_i are the normal Breit-Wigner parameters, and $H_i^{(r)}$ describes the interference added to the cross-sections of a given level from all the adjacent ones.

From the experimental data cited above, it is easy to obtain the values of $G_i^{(r)}$ and $H_i^{(r)}$ for the (nT), (nf) and (nγ) reactions and also the values of Er and Γ for all investigated levels by the method of least squares. The quality of parametrization was determined by two criteria: (1) reproduction in detail of the shape of the curve σ_{nr} (E); (2) agreement of the mean cross-sections from the parameters with the experimental or evaluated mean cross-sections in the same energy ranges.

As a result of parametrizing the cross-sections σ_T, σ_f and σ_γ, three sets of parameters were obtained that describe the experimental data satisfactorily in terms of the corresponding cross-section types:

(1) G^T, H^T, Γ, E_r (2) G^f, H^f, Γ, E_r; (3) G^γ, H^γ, Γ, E_r for 208 resonances. After this we set ourselves the task of making these quantities agree and obtaining self-consistant Breit-Wigner parameters from them. Agreement was difficult because we did not parametrize the cross-section σ_n, owing to the poor quality of the experimental data; besides, there were no reliable data on the spin of most of the levels. The agreement procedure was as follows:

(1) Determination of the spins of those levels for which they had not been known (above 58.7 eV). It is well-known that

$$G_i^T = G_i^f + G_i^\gamma + G_i^n \text{ , but } G_i^n = \frac{G_i^{T2}}{g} \text{ ,}$$

so we checked at which of the two values of g (7/16 or 9/16, corresponding to J = 3 or 4) the difference $G_i^T - G_i^f - G_i^\gamma - G_i^n$ is smallest. It should be noted that the values obtained for J are not very reliable, as determination of J without taking the data on σ_n into account does not afford sufficient certainty that the spins of the levels have been determined correctly.

(2) For the parameters to agree, the relation $\hat{G}_i^T = \hat{G}_i^f + \hat{G}_i^\gamma + \hat{G}_i^n$ must hold; it was therefore necessary to find increments to the parameters with which the condition would still be satisfied while the sum of the squares of these increments had to be as small as possible.

(3) The self-consistent values of the Breit-Wigner parameters were calculated from the following expressions:

$$\Gamma_n = \frac{\hat{G}^T \Gamma}{g} \;; \quad \Gamma_f = \frac{\hat{G}^f \Gamma}{\hat{G}^T} \;; \quad \Gamma_\gamma = \frac{\hat{G}^\gamma \Gamma}{\hat{G}^T}$$

Limits were placed only on the quantity Γ_γ. Because of the insufficiently reliable data on σ_γ, anomalously high (above 0.07 eV) or anomalously low (below 0.01 eV) values of Γ_γ were obtained at some levels, for which reason the value of Γ_γ was modified until it fell within the 0.01-0.07 eV range; after this the remaining parameters were corrected. Tables of the self-consistent resonance parameters of both types have been sent to the Obninsk Nuclear Data Centre.

The following values for the mean resonance parameters were obtained as a result of the analysis:

$\langle D\rangle = 0.610 \pm 0.051$ eV, $\langle \Gamma\rangle = 173.2 \pm 5.0$ meV,

$\langle \Gamma_f\rangle = 131.4 \pm 5.2$ meV, $\langle \Gamma_\gamma\rangle = 40.69 \pm 2.0$ meV.

The value of $\langle \Gamma_\gamma\rangle$ agrees, within the margin of error, with the one obtained from the paper by Smith et al. [16]. The value of $\langle \Gamma_f\rangle$ agrees with the data of Krebs et al. [17]. It should also be noted that the value $S_0 = (1.069 \pm 0.14 \times 10^{-4}$ derived from the parameters described above, agrees well with the value $S_0 = 1.08 \times 10^{-4}$ obtained from the values of σ_T by an independent method for the unresolved region.

Use of the self-consistent parameters permitted the cross-sections σ_T, σ_f and σ_γ to be established in detail. All types of cross-section are satisfactorily described in the investigated energy region. It should be pointed out that each type of cross-section can be described separately with a sufficiently high degree of accuracy, although some deterioration in the representation of one or the other type of cross-section for the individual resonances is found in the self-consistency procedure. The calculated cross-section values, averaged over a sufficiently narrow range (~ 20 eV), agree with the experimental values of σ_f with an accuracy of ~ 5%, and with those of α with an accuracy of ~ 10%.

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TABLE 1

R-matrix resonance parameters for ^{235}U

	E_r	Γ	Γ_n	Γ_f	Γ_γ	J
1	-1.49000E-00	2.37680E-01	3.26879E-03	2.12253E-01	2.22583E-02	4
2	2.90000E-01	1.35000E-01	3.44756E-06	1.00738E-01	3.42589E-02	3
3	1.14000E-00	1.50820E-01	1.24136E-05	1.15254E-01	3.55540E-02	4
4	2.03650E-00	4.46960E-02	8.44055E-06	1.01641E-02	3.45235E-02	3
5	2.92000E-00	2.20000E-01	6.47719E-06	2.09120E-01	1.08733E-02	4
6	3.15640E-00	1.39610E-01	3.02943E-05	1.07002E-01	3.25773E-02	3
7	3.62570E-00	8.43790E-02	4.16105E-05	4.95037E-02	3.48337E-02	4
8	4.85010E-00	3.95920E-02	5.49558E-05	3.24543E-03	3.62916E-02	4
9	5.51080E-00	7.31000E-01	5.68309E-05	6.71776E-01	5.92674E-02	4
10	6.21060E-00	2.30900E-01	9.19167E-05	1.65618E-01	6.51901E-02	3
11	6.38170E-00	4.47880E-02	2.19179E-04	9.56905E-03	3.49998E-02	4
12	7.07720E-00	6.39340E-02	1.09054E-04	2.82378E-02	3.55872E-02	4
13	8.78970E-00	1.23290E-01	1.02880E-03	7.53982E-02	4.68630E-02	4
14	9.28160E-00	1.11000E-01	1.29176E-04	6.92646E-02	4.16063E-02	4
15	9.74890E-00	2.69050E-01	6.09756E-05	2.21759E-01	4.72301E-02	3
16	1.01820E 01	1.00560E-01	6.11065E-05	5.47474E-02	4.57515E-02	4
17	1.08000E 01	9.35090E-01	1.01106E-04	8.65104E-01	6.98846E-02	4
18	1.16660E 01	4.72780E-02	5.48862E-04	5.52260E-03	4.12065E-02	4
19	1.24040E 01	6.32620E-02	1.48170E-03	2.52850E-02	3.64953E-02	3
20	1.28610E 01	1.19550E-01	7.22847E-05	6.39737E-02	5.55040E-02	4
21	1.32740E 01	1.51440E-01	4.91016E-05	1.31794E-01	1.95971E-02	4
22	1.37000E 01	1.23940E-01	2.95586E-05	8.62805E-02	3.76300E-02	3
23	1.39960E 01	4.96540E-01	5.81094E-04	4.31631E-01	6.43279E-02	3
24	1.45500E 01	5.62150E-02	1.44412E-04	1.73852E-02	3.86854E-02	3

25	1.54000E 01	8.50000E-02	2.19141E-04	4.84662E-02	3.63147E-02	4
26	1.60950E 01	5.60000E-02	3.33322E-04	2.24997E-02	4.31670E-02	4
27	1.66610E 01	1.20000E-01	2.40427E-04	8.74296E-02	3.23299E-02	4
28	1.80630E 01	1.55000E-01	4.23974E-04	1.16631E-01	3.79446E-02	3
29	1.89610E 01	1.00000E-01	1.02674E-04	6.20426E-02	3.78549E-02	4
30	1.93010E 01	1.05000E-01	3.01168E-03	6.61017E-02	3.58866E-02	4
31	2.01300E 01	2.20000E-01	1.80826E-04	1.52126E-01	9.76930E-02	4
32	2.06330E 01	9.10000E-02	1.85640E-04	6.96834E-02	2.11310E-02	4
33	2.10610E 01	6.90000E-02	1.35019E-03	2.85021E-02	3.91477E-02	4
34	2.29610E 01	9.00000E-02	4.43744E-04	5.05686E-02	3.89876E-02	4
35	2.34110E 01	5.00000E-02	6.60596E-04	1.55980E-02	3.37414E-02	4
36	2.36100E 01	2.25860E-01	8.72413E-04	1.93476E-01	3.15117E-02	3
37	2.42700E 01	7.50000E-02	3.67766E-04	3.00371E-02	4.45952E-02	3
38	2.44000E 01	1.00150E-01	2.15576E-04	3.34397E-02	9.65147E-02	4
39	2.52300E 01	8.50680E-01	8.83467E-04	7.82050E-01	6.77470E-02	4
40	2.55660E 01	3.85560E-01	7.62757E-04	3.74441E-01	1.03562E-02	3
41	2.64700E 01	1.90000E-01	5.69866E-04	1.66342E-01	2.30881E-02	3
42	2.67900E 01	2.50090E-01	7.56272E-05	2.15497E-01	3.45173E-02	3
43	2.71500E 01	1.44000E-01	8.75418E-05	1.10367E-01	3.35459E-02	4
44	2.77990E 01	1.28000E-01	7.93145E-04	6.55342E-02	6.16626E-02	4
45	2.80900E 01	6.50310E-02	6.26749E-05	5.40945E-02	1.08738E-02	4
46	2.83510E 01	1.49190E-01	1.95642E-04	8.28631E-02	6.61313E-02	4
47	2.87090E 01	1.30040E-01	6.51121E-05	6.09002E-02	6.90747E-02	3
48	2.96490E 01	5.11770E-02	1.61337E-04	2.19253E-02	2.90903E-02	4
49	3.05950E 01	1.55230E-01	2.74862E-04	9.22719E-02	6.26832E-02	3
50	3.08620E 01	5.10320E-02	5.46192E-04	2.43242E-02	2.61616E-02	4
51	3.20730E 01	9.10230E-02	1.92548E-03	5.84157E-02	3.06819E-02	4
52	3.35200E 01	5.10590E-02	1.93189E-03	2.03986E-02	2.87285E-02	4
53	3.43710E 01	8.72530E-02	2.76216E-03	3.86830E-02	4.58079E-02	4
54	3.48430E 01	1.16100E-01	1.88493E-03	7.98937E-02	3.43213E-02	3
55	3.51870E 01	1.03500E-01	3.99206E-03	6.75498E-02	3.19582E-02	4
56	3.64000E 01	1.54010E-00	5.59540E-04	1.47107E-00	6.84739E-02	3

57	3.75030E 01	1.54020E-00	2.30523E-04	1.52169E-00	1.22833E-02	4
58	3.82990E 01	3.08340E-01	3.01840E-04	2.69392E-01	3.86471E-02	4
59	3.94140E 01	9.10230E-02	2.82519E-03	5.69734E-02	3.12244E-02	4
60	3.99000E 01	1.50240E-01	3.28872E-04	1.27240E-01	2.26714E-02	4
61	4.05360E 01	2.09380E-01	4.24418E-04	1.58561E-01	5.03946E-02	4
62	4.13650E 01	4.45640E-01	5.43871E-04	3.77641E-01	6.74554E-02	4
63	4.16140E 01	1.65220E-01	2.60292E-04	1.20286E-01	4.46733E-02	3
64	4.18740E 01	4.12330E-02	2.27823E-03	7.68865E-03	3.12661E-02	3
65	4.22300E 01	1.45450E-01	4.54011E-04	7.82148E-02	6.67812E-02	4
66	4.26640E 01	6.13450E-02	3.83763E-04	1.93367E-02	4.16245E-02	4
67	4.34050E 01	7.07540E-02	5.81353E-04	2.05811E-02	4.95916E-02	4
68	4.39320E 01	3.60540E-01	1.27641E-03	2.92076E-01	6.71874E-02	4
69	4.46000E 01	1.75840E-01	9.56851E-04	1.19450E-01	5.54333E-02	4
70	4.49500E 01	5.35760E-01	2.67831E-04	5.13836E-01	2.16564E-02	3
71	4.58570E 01	1.34190E-01	1.35113E-04	1.23322E-01	1.07329E-02	4
72	4.67900E 01	1.52800E-01	6.68816E-04	1.30098E-01	2.20830E-02	4
73	4.70150E 01	1.39940E-01	1.02988E-03	8.21007E-02	5.68094E-02	4
74	4.79700E 01	9.39880E-02	7.47409E-04	4.69719E-02	4.62687E-02	4
75	4.83000E 01	1.65770E-01	1.22181E-03	1.16485E-01	4.80630E-02	3
76	4.87390E 01	6.56910E-02	7.32571E-04	4.23479E-02	2.26105E-02	3
77	4.94180E 01	6.10130E-02	8.61504E-04	1.75238E-02	4.26277E-02	4
78	5.01070E 01	5.43530E-02	2.28466E-04	3.07052E-02	2.34194E-02	4
79	5.04660E 01	7.59640E-02	1.00318E-03	4.78272E-02	2.71336E-02	3
80	5.07800E 01	3.30190E-01	3.74839E-04	2.64020E-01	6.57948E-02	3
81	5.12580E 01	1.88540E-01	3.21355E-03	1.24100E-01	6.12260E-02	4
82	5.16300E 01	7.43460E-02	4.76970E-04	1.59984E-02	5.78707E-02	3
83	5.21650E 01	6.63510E-01	4.89981E-03	5.96847E-01	6.17634E-02	3
84	5.34400E 01	1.35540E-01	5.17413E-04	9.79261E-02	3.70965E-02	4
85	5.41190E 01	1.50210E-01	2.00189E-04	1.28897E-01	2.11133E-02	4
86	5.50640E 01	1.11170E-01	2.40602E-03	6.42246E-02	4.45394E-02	4
87	5.58460E 01	2.51350E-01	2.05834E-03	2.29540E-01	1.97514E-02	4
88	5.60770E 01	1.80770E-01	5.51501E-04	1.69699E-01	1.05195E-02	4

89	5.65040E 01	1.19920E-01	3.98262E-03	7.59988E-02	3.99386E-02	4
90	5.78210E 01	2.21130E-01	1.11891E-03	1.76139E-01	4.38769E-02	4
91	5.80580E 01	6.53540E-02	1.61749E-03	2.90399E-02	3.46966E-02	3
92	5.86800E 01	1.36830E-01	1.16917E-03	9.58295E-02	3.95314E-02	4
93	5.97810E 01	2.55270E-01	2.86297E-04	2.02684E-01	5.22996E-02	4
94	6.01860E 01	2.55130E-01	1.37041E-03	2.28933E-01	2.48266E-02	3
95	6.08410E 01	1.20460E-01	3.95216E-04	9.69163E-02	2.31485E-02	4
96	6.11250E 01	1.25360E-01	4.34906E-04	7.59280E-02	4.89971E-02	3
97	6.16440E 01	5.30230E-01	2.95562E-04	5.15805E-01	1.41291E-02	4
98	6.19000E 01	5.30170E-01	3.41863E-05	4.78075E-01	5.20605E-02	4
99	6.24000E 01	5.16240E-01	2.89425E-04	4.87314E-01	2.86361E-02	4
100	6.30200E 01	2.40090E-01	6.82325E-05	1.77567E-01	6.24549E-02	4
101	6.33200E 01	4.12540E-02	3.38701E-05	1.00799E-03	4.02121E-02	4
102	6.36880E 01	6.21070E-01	9.54957E-04	5.52371E-01	6.77440E-02	4
103	6.42930E 01	4.75450E-02	1.28866E-03	8.17944E-03	3.80769E-02	3
104	6.57900E 01	9.64230E-02	3.51648E-04	3.40019E-02	6.20695E-02	4
105	6.64040E 01	8.94490E-02	3.65302E-04	6.07009E-02	2.83828E-02	4
106	6.72600E 01	9.00810E-02	7.37079E-05	3.91925E-02	5.08148E-02	4
107	6.84000E 01	2.50040E-01	7.48209E-05	1.93012E-01	5.69537E-02	4
108	6.85300E 01	1.60110E-01	9.55622E-05	9.34869E-02	6.65275E-02	4
109	6.92930E 01	2.00720E-01	7.44293E-04	1.59618E-01	4.03573E-02	3
110	7.04040E 01	1.72720E-01	2.29225E-03	1.24489E-01	4.59385E-02	4
111	7.07500E 01	2.37410E-01	2.19515E-03	1.76439E-01	5.87755E-02	4
112	7.15700E 01	3.21020E-02	3.97808E-04	2.14734E-02	1.02308E-02	3
113	7.23900E 01	1.38610E-01	2.65047E-03	8.15204E-02	5.44392E-02	4
114	7.28750E 01	3.60370E-01	4.13804E-04	2.94690E-01	6.52658E-02	3
115	7.45440E 01	1.01670E-01	2.49358E-03	6.38356E-02	3.53408E-02	4
116	7.51800E 01	2.90890E-01	6.43113E-04	2.23924E-01	6.63229E-02	4
117	7.55410E 01	2.33360E-01	1.10196E-03	2.11651E-01	2.06070E-02	4
118	7.67540E 01	1.16110E-01	9.83622E-05	7.74335E-02	3.85782E-02	4
119	7.74880E 01	1.12990E-01	9.52631E-04	8.18603E-02	3.01771E-02	4

120	7.81200E 01	1,48220E-01	1.14969E-03	1,02749E-01	4,43208E-02	4
121	7.84190E 01	1.78780E-01	2.08049E-04	1,09389E-01	6,91829E-02	4
122	7.96760E 01	1.29790E-01	8.67116E-04	8.77078E-02	4,12151E-02	3
123	8.03570E 01	1,74840E-01	9.31627E-04	1,31175E-01	4,27334E-02	3
124	8.14350E 01	1,32040E-01	9.01651E-04	8,83452E-02	4,27931E-02	4
125	8.27060E 01	6,49000E-02	1.45830E-03	1,94371E-02	5,40046E-02	3
126	8.35920E 01	1,18270E-01	1.44771E-03	6,55539E-02	5,12684E-02	3
127	8.40600E 01	1,15000E-01	1,97992E-03	9,87278E-02	1,42923E-02	3
128	8.43600E 01	1,31000E-01	1,73859E-03	8,89711E-02	4,02903E-02	4
129	8,50100E 01	1,50000E-01	1,18443E-03	1,30413E-01	1,84025E-02	3
130	8,55660E 01	1,37000E-01	5,88270E-04	1,16887E-01	1,95249E-02	3
131	8.61410E 01	1,59200E-01	9.58285E-05	8,93305E-02	6,97737E-02	4
132	8,68740E 01	8,81200E-02	5,19374E-04	5,72271E-02	3,03735E-02	3
133	8,75130E 01	1,90800E-01	5,74537E-04	1,59676E-01	3,05496E-02	4
134	8,84300E 01	1,78800E-01	8,04775E-04	1,67395E-01	1,06006E-02	4
135	8,87500E 01	1,47900E-01	1,28490E-03	1,10190E-01	3,64255E-02	4
136	8,91140E 01	1,40300E-01	5,43366E-04	7,35021E-02	6,62545E-02	4
137	8,98040E 01	1,30000E-01	5,97515E-04	9,67483E-02	3,26542E-02	4
138	9,04090E 01	5,80000E-02	4,09764E-03	1,08326E-02	4,30698E-02	4
139	9,12890E 01	2,80000E-01	3,12326E-03	2,34948E-01	4,19283E-02	3
140	9,20700E 01	1,20000E-01	7,53088E-04	6,18509E-02	5,73960E-02	4
141	9,25900E 01	9,00000E-02	2,38224E-03	3,83804E-02	4,92373E-02	4
142	9,32270E 01	1,20000E-01	5,54825E-04	5,01250E-02	6,93202E-02	3
143	9,41140E 01	7,50000E-02	3,58307E-03	8,73832E-03	6,26786E-02	4
144	9,47580E 01	1,05000E-01	5,68736E-04	4,62830E-02	5,81483E-02	4
145	9,55000E 01	8,64470E-02	2,34879E-04	7,18862E-02	1,43259E-02	3
146	9,55690E 01	1,52270E-01	7,57521E-04	1,41132E-01	1,03803E-02	3
147	9,60900E 01	1,50000E-01	3,63013E-04	8,30151E-02	9,66219E-02	4
148	9,65000E 01	2,60100E-01	7,09414E-04	2,49012E-01	1,03782E-02	4
149	9,81300E 01	2,30000E-01	3,08121E-03	1,91264E-01	3,56550E-02	3
150	9,95180E 01	2,25000E-01	6,78189E-04	1,90688E-01	3,36340E-02	3
151	1,00360E 02	1,30000E-01	5,55730E-04	8,78146E-02	4,16297E-02	4

152	1.01000E 02	8.00000E-02	8.63886E-04	3.94705E-02	3.96656E-02	4
153	1.01860E 02	7.50000E-02	3.34600E-04	3.81052E-02	3.65602E-02	4
154	1.02940E 02	1.30000E-01	2.36734E-03	9.91472E-02	2.84854E-02	3
155	1.03570E 02	1.46000E-01	2.02539E-03	7.85298E-02	6.54449E-02	4
156	1.04200E 02	8.77000E-02	3.61028E-04	4.44863E-02	4.28527E-02	4
157	1.05150E 02	1.24000E-01	1.79766E-03	9.69516E-02	2.52507E-02	4
158	1.05440E 02	8.65000E-02	4.49416E-04	5.44069E-02	3.16436E-02	4
159	1.06120E 02	1.06800E-01	8.08281E-04	6.42416E-02	4.17502E-02	4
160	1.06720E 02	1.20600E-01	1.28286E-04	9.93038E-02	2.11679E-02	4
161	1.07670E 02	7.10000E-02	3.50898E-03	1.69463E-02	5.05448E-02	4
162	1.07990E 02	5.78000E-02	5.61737E-04	3.40297E-02	2.32086E-02	3
163	1.08920E 02	9.70000E-02	1.59304E-03	3.46873E-02	6.07196E-02	3
164	1.09830E 02	1.00000E-01	2.05156E-03	3.11265E-02	6.68219E-02	4
165	1.10140E 02	6.19000E-02	5.97395E-04	5.05955E-02	1.07071E-02	3
166	1.11170E 02	1.03000E-01	4.48494E-04	3.45514E-02	6.80001E-02	4
167	1.11690E 02	9.80000E-02	1.15650E-03	3.15535E-02	6.52900E-02	4
168	1.12860E 02	1.30000E-01	6.97724E-04	6.59473E-02	6.33549E-02	4
169	1.13570E 02	2.03000E-01	1.75188E-03	1.46223E-01	5.50253E-02	3
170	1.15130E 02	5.50000E-02	5.92881E-04	2.30246E-02	3.13825E-02	3
171	1.15980E 02	1.18300E-01	2.37744E-03	9.20009E-02	2.39216E-02	3
172	1.16510E 02	1.78780E-01	3.38808E-04	1.68013E-01	1.04282E-02	4
173	1.17410E 02	1.78780E-01	2.00806E-04	1.36343E-01	4.22361E-02	4
174	1.18240E 02	1.49200E-01	2.18743E-03	1.29096E-01	1.79170E-02	3
175	1.18690E 02	1.10000E-01	2.53538E-03	4.93460E-02	5.81186E-02	4
176	1.19500E 02	1.78780E-01	4.24877E-04	1.52688E-01	2.56673E-02	4
177	1.20300E 02	2.00000E-01	3.55317E-04	1.43918E-01	5.57263E-02	4
178	1.21010E 02	2.00000E-01	1.09849E-03	1.51714E-01	4.71876E-02	4
179	1.21950E 02	1.50000E-01	5.46613E-03	1.07857E-01	3.66770E-02	4
180	1.22900E 02	1.15100E-01	6.08874E-04	4.91639E-02	6.53272E-02	4
181	1.23570E 02	9.00000E-02	2.94656E-04	7.61745E-02	1.35308E-02	4
182	1.23960E 02	1.20000E-01	1.64757E-04	5.61497E-02	6.36856E-02	4
183	1.24780E 02	1.82000E-01	2.14475E-03	1.84572E-01	5.52830E-02	4

184	1,25600E 02	8.73000E-02	2.87679E-03	2.55485E-02	5.88747E-02	4
185	1,26000E 02	1,50000E-01	3,58834E-03	1,13270E-01	3,31420E-02	3
186	1,26450E 02	8,90000E-02	2,82257E-03	7,08087E-02	1,53687E-02	3
187	1,27710E 02	6,21850E-02	4,58046E-04	5,10111E-02	1,07158E-02	4
188	1,28190E 02	2,30000E-01	1,41709E-03	1,64073E-01	6,45096E-02	4
189	1,29560E 02	1,20000E-01	8,86217E-04	8,96045E-02	2,95093E-02	3
190	1,29930E 02	6,40000E-02	1,63055E-03	1,19046E-02	5,04649E-02	4
191	1,31240E 02	1,55000E-01	1,86132E-03	1,11679E-01	4,14596E-02	4
192	1,31640E 02	2,00000E-01	1,32668E-03	1,87908E-01	1,07656E-02	4
193	1,32140E 02	1,28700E-01	1,27032E-03	1,04534E-01	2,28953E-02	4
194	1,32700E 02	7,50000E-02	1,39200E-03	6,32881E-02	1,03199E-02	4
195	1,33030E 02	1,12000E-01	8,25455E-04	7,33402E-02	3,78344E-02	4
196	1,33620E 02	7,00000E-02	3,98367E-03	2,57651E-02	4,02515E-02	4
197	1,35080E 02	2,00000E-01	3,70098E-03	1,47071E-01	4,92282E-02	4
198	1,35470E 02	2,50000E-01	4,41806E-03	2,19263E-01	2,68194E-02	3
199	1,36360E 02	9,00000E-02	2,75200E-03	3,05717E-02	5,66763E-02	4
200	1,37530E 02	5,90000E-02	2,96521E-03	2,15046E-02	3,45302E-02	4
201	1,39190E 02	4,50000E-02	5,14024E-04	2,04665E-02	2,40195E-02	4
202	1,40240E 02	1,06200E-01	8,52375E-04	6,46275E-02	4,07201E-02	4
203	1,41810E 02	1,00000E-01	1,26492E-03	6,25970E-02	3,61381E-02	4
204	1,42140E 02	9,50000E-02	4,12869E-03	4,04250E-02	5,04465E-02	4
205	1,43050E 02	9,33000E-02	2,03684E-04	8,21314E-02	1,09650E-02	4
206	1,45600E 02	1,05100E-01	4,74940E-03	5,10636E-02	4,92870E-02	4
207	1,46350E 02	1,20000E-01	4,77760E-04	5,60055E-02	6,35169E-02	4
208	1,47350E 02	7,30000E-02	2,37818E-03	3,67163E-02	3,39055E-02	4

TABLE 2

Adler-Adler resonance parameters for ^{235}U

	E_r	Γ_T	C_T	H_T	G_f	H_f	G_γ	H_γ	J
1	-1.49000E-00	2.37680E-01	7.73600E-03	2.00000E-06	6.90840E-03	1.00000E-07	7.21208E-04	6.00000E-08	4
2	2.90000E-01	1.35000E-01	1.11720E-05	1.00000E-06	8.33660E-06	5.00000E-08	2.83511E-06	3.00000E-08	3
3	1.14000E-00	1.50820E-01	4.62980E-05	-7.74590E-06	3.53800E-05	-1.00910E-05	1.09142E-05	-8.92570E-07	4
4	2.03650E-00	4.46960E-02	8.26190E-05	-1.94530E-06	1.87880E-05	-3.98340E-07	6.38154E-05	-1.57150E-06	3
5	2.92000E-00	2.20000E-01	1.65610E-05	-3.32590E-06	1.57420E-05	-1.33120E-06	8.18512E-07	0.00000E-01	4
6	3.15640E-00	1.39610E-01	9.49340E-05	-1.26310E-06	7.27610E-05	-2.84900E-06	2.21524E-05	2.02070E-06	3
7	3.62570E-00	8.43790E-02	2.77390E-04	-9.49760E-06	1.62740E-04	-9.94300E-06	1.14513E-04	-3.34880E-06	4
8	4.85010E-00	3.95920E-02	7.80780E-04	-4.16300E-05	6.40020E-05	1.12830E-06	7.15694E-04	-1.81100E-05	4
9	5.51080E-00	7.31000E-01	4.37310E-05	5.72350E-06	4.01880E-05	6.69210E-06	3.53960E-06	-3.05990E-06	4
10	6.21060E-00	2.30900E-01	1.74160E-04	3.54040E-06	1.24920E-04	-9.77200E-16	4.91707E-05	2.12380E-06	3
11	6.38170E-00	4.47860E-02	2.75270E-03	1.44210E-05	5.88120E-04	-1.44150E-05	2.15111E-03	-8.46100E-06	4
12	7.07720E-00	6.39340E-02	9.59470E-04	-2.45370E-05	4.23770E-04	6.61410E-06	5.34063E-04	-3.36700E-05	4
13	8.78970E-00	1.23290E-01	4.69380E-03	1.51920E-04	2.87050E-03	5.67370E-05	1.78413E-03	2.61330E-05	4
14	9.28160E-00	1.11000E-01	6.54610E-04	-6.70890E-05	4.08480E-04	-2.88520E-05	2.45368E-04	-4.09050E-05	4
15	9.74890E-00	2.69050E-01	9.91520E-05	-2.95060E-05	8.17240E-05	1.75310E-05	1.74055E-05	9.70080E-08	3
16	1.01820E 01	1.00560E-01	3.41810E-04	-2.78960E-05	1.86090E-04	1.32640E-05	1.55512E-04	-2.99180E-06	4
17	1.08000E 01	9.35090E-01	6.08200E-05	-5.01910E-06	5.62680E-05	-1.33680E-06	4.54542E-06	0.00000E-01	4
18	1.16660E 01	4.72780E-02	6.53020E-03	-8.29380E-05	7.62800E-04	-1.76420E-05	5.69159E-03	7.18330E-05	4
19	1.24040E 01	6.32620E-02	1.02470E-02	-2.87300E-04	4.09560E-03	-1.34830E-04	5.91140E-03	0.00000E-01	3
20	1.28610E 01	1.19550E-01	3.40110E-04	1.49910E-05	1.82000E-04	-3.82010E-05	1.57904E-04	-4.09940E-14	4
21	1.32740E 01	1.51440E-01	1.82380E-04	2.00750E-06	1.58720E-04	-1.36120E-05	2.36009E-05	1.24150E-10	4

22	1.37000E 01	1.23940E-01	1.04840E-04	3.47720E-05	7.26360E-05	5.62840E-05	3.16791E-05	2.18190E-11	3
23	1.39960E 01	4.96540E-01	5.12000E-04	-1.27560E-05	4.45070E-04	1.31840E-06	6.63308E-05	-1.18350E-14	3
24	1.45500E 01	5.62150E-02	1.12390E-03	-1.27800E-04	3.47580E-04	-6.14740E-05	7.73433E-04	3.23580E-06	3
25	1.54000E 01	8.50000E-02	1.45020E-03	2.10710E-05	8.26890E-04	-5.21460E-05	6.19571E-04	2.29010E-06	4
26	1.60950E 01	5.60000E-02	3.34810E-03	-3.97570E-04	1.34520E-03	-5.24240E-05	1.98297E-03	-3.34370E-06	4
27	1.66610E 01	1.20000E-01	1.12700E-03	5.14860E-05	8.21110E-04	5.13520E-05	3.03632E-04	6.35810E-06	4
28	1.80630E 01	1.55000E-01	1.19670E-03	1.51020E-05	9.00470E-04	6.66000E-06	2.92957E-04	-2.27620E-06	3
29	1.89610E 01	1.00000E-01	5.77540E-04	-4.22840E-06	3.58320E-04	-3.83430E-06	2.18627E-04	4.18840E-06	4
30	1.93010E 01	1.05000E-01	1.61340E-02	-1.21410E-03	1.01570E-02	1.01320E-03	5.51423E-03	1.53570E-04	4
31	2.01300E 01	2.20000E-01	4.62340E-04	0.00000E-01	3.19700E-04	3.18370E-05	1.42260E-04	0.00000E-01	4
32	2.06330E 01	9.10000E-02	1.14750E-03	-3.91320E-05	8.78700E-04	7.32820E-05	2.66459E-04	0.00000E-01	4
33	2.10610E 01	6.90000E-02	1.10070E-02	5.80580E-04	5.54670E-03	-1.66800E-04	6.24491E-03	-4.44930E-06	4
34	2.29610E 01	9.00000E-02	2.77340E-03	-1.14300E-04	1.55830E-03	-1.19770E-04	1.20143E-03	0.00000E-01	4
35	2.34110E 01	5.00000E-02	7.43170E-03	1.17080E-04	2.31840E-03	-2.40630E-05	5.01511E-03	1.18260E-04	4
36	2.36100E 01	2.25860E-01	1.68990E-03	-4.38170E-05	1.44760E-03	1.04850E-04	2.35773E-04	-5.35360E-05	3
37	2.42700E 01	7.50000E-02	2.14530E-03	-8.10010E-04	8.59180E-04	-1.29090E-04	1.27560E-03	-1.31980E-06	3
38	2.44000E 01	1.00150E-01	1.21080E-03	9.80350E-05	4.04040E-04	-1.10940E-04	8.04154E-04	2.86540E-14	4
39	2.52300E 01	8.50680E-01	5.84180E-04	-5.99750E-05	5.37050E-04	-3.45410E-05	4.65233E-05	0.00000E-01	4
40	2.55660E 01	3.85560E-01	8.65510E-04	7.05910E-05	8.40550E-04	-4.07000E-05	2.32478E-05	0.00000E-01	3
41	2.64700E 01	1.90000E-01	1.31150E-03	2.28310E-04	1.14820E-03	1.63250E-04	1.59368E-04	0.00000E-01	3
42	2.67900E 01	2.50090E-01	1.32300E-04	-3.07580E-06	1.14000E-04	-3.68430E-05	1.82600E-05	0.00000E-01	3
43	2.71500E 01	1.44000E-01	3.41960E-04	-9.98240E-05	2.62090E-04	2.62990E-06	7.96621E-05	0.00000E-01	4
44	2.77990E 01	1.28000E-01	3.48550E-03	-1.85620E-04	1.78480E-03	5.70360E-05	1.67910E-03	0.00000E-01	4

45	2.80900E 01	6.50310E-02	5.42120E-04	1.00340E-04	4.50950E-04	2.37790E-05	9.06475E-05	0.00000E-01	4
46	2.83510E 01	1.49190E-01	7.37640E-04	4.84080E-05	4.09700E-04	1.03020E-04	3.26973E-04	4.17530E-06	4
47	2.87090E 01	1.30040E-01	2.19060E-04	1.43060E-05	1.02590E-04	-5.25780E-05	1.16360E-04	1.78790E-06	3
48	2.96490E 01	5.11770E-02	1.77330E-03	-3.70900E-06	7.59720E-04	-2.91250E-05	1.00799E-03	-5.60630E-05	4
49	3.05950E 01	1.55230E-01	7.74670E-04	6.84290E-05	4.60480E-04	3.73790E-05	3.12818E-04	2.24590E-05	3
50	3.08620E 01	5.10320E-02	6.02040E-03	-2.71000E-04	2.86960E-03	-9.24650E-05	3.08636E-03	-8.45490E-05	4
51	3.20730E 01	9.10230E-02	1.18990E-02	-2.76740E-04	7.63640E-03	1.47730E-05	4.01089E-03	-2.24160E-05	4
52	3.35200E 01	5.10590E-02	2.12830E-02	-2.20640E-04	8.50280E-03	7.44880E-05	1.19749E-02	-4.01480E-04	4
~~53~~	~~3.43710E 01~~	~~9.72530E-02~~	~~1.78070E-02~~	~~-4.69090E-04~~	~~7.89460E-03~~	~~1.49890E-04~~	~~9.34869E-03~~	~~-3.15400E-04~~	~~4~~
54	3.48430E 01	1.16100E-01	7.10300E-03	3.73180E-04	4.88790E-03	3.67410E-04	2.09978E-03	3.76950E-04	3
55	3.51870E 01	1.03500E-01	2.16960E-02	9.62960E-04	1.41600E-02	1.12720E-03	6.69917E-03	-2.45790E-04	4
56	3.64000E 01	1.54010E-00	1.58950E-04	-5.94610E-05	1.52825E-04	-4.03620E-05	7.06703E-06	-2.82490E-09	3
57	3.75030E 01	1.54020E-00	8.41900E-05	4.31550E-06	8.31780E-05	4.62180E-05	9.99400E-07	4.89210E-10	4
58	3.82990E 01	3.08340E-01	5.49730E-04	-9.13120E-06	4.80290E-04	-6.68050E-05	6.89028E-05	6.37790E-06	4
59	3.94140E 01	9.10230E-02	1.74590E-02	-6.15120E-04	1.09280E-02	-2.29900E-05	5.98910E-03	-8.12630E-05	4
60	3.99000E 01	1.50240E-01	1.23130E-03	-2.45130E-04	1.04280E-03	2.80680E-06	1.85805E-04	0.00000E-01	4
61	4.05360E 01	2.09380E-01	1.14020E-03	0.00000E-01	8.63460E-04	7.08600E-05	2.74429E-04	0.00000E-01	4
62	4.13650E 01	4.45640E-01	6.86490E-04	-2.53440E-11	5.81740E-04	1.75610E-04	1.03912E-04	4.55310E-05	4
63	4.16140E 01	1.65220E-01	6.89250E-04	-6.17280E-13	5.01800E-04	-2.66480E-04	1.86364E-04	9.86210E-05	3
64	4.18740E 01	4.12330E-02	2.41730E-02	-1.13000E-13	4.50750E-03	2.74730E-04	1.83299E-02	-1.04280E-03	3
65	4.22300E 01	1.45450E-01	1.75580E-03	3.38540E-14	9.44170E-04	1.74680E-05	8.06149E-04	-3.16350E-06	4
66	4.26640E 01	6.13450E-02	3.51890E-03	1.86590E-13	1.10920E-03	2.06540E-05	2.38769E-03	3.13940E-04	4
67	4.34050E 01	7.07540E-02	4.62180E-03	-4.05780E-14	1.34440E-03	-9.60660E-05	3.23942E-03	-3.26420E-04	4

68	4.39320E 01	3.60540E-01	1.99140E-03	8.94230E-10	1.62325E-03	-7.49350E-05	3.71101E-04	1.23510E-10	4
69	4.46000E 01	1.75840E-01	3.06090E-03	7.30180E-04	2.07930E-03	-2.54120E-04	9.64944E-04	-2.16150E-14	4
70	4.49500E 01	5.35760E-01	2.18710E-04	-6.05120E-05	2.09760E-04	2.05180E-05	8.84066E-06	-3.74960E-12	3
71	4.58570E 01	1.34190E-01	5.66370E-04	-1.81810E-04	5.20500E-04	-1.77370E-05	4.52997E-05	0.00000E-01	4
72	4.67900E 01	1.52800E-01	2.46210E-03	7.77890E-05	2.09630E-03	1.55350E-04	3.55023E-04	-2.10090E-14	4
73	4.70150E 01	1.39940E-01	4.13970E-03	-1.14280E-04	2.42870E-03	-1.28680E-04	1.68053E-03	6.07520E-07	4
74	4.79700E 01	9.39880E-02	4.47310E-03	-8.01980E-04	2.23550E-03	-1.55240E-04	2.20203E-03	5.22020E-05	4
75	4.83000E 01	1.65770E-01	3.22460E-03	4.18820E-04	2.26590E-03	1.27460E-04	9.34933E-04	1.70910E-05	3
76	4.87390E 01	6.56910E-02	4.87890E-03	1.61130E-03	3.14520E-03	6.43450E-04	1.67929E-03	1.64330E-05	3
77	4.94180E 01	6.10130E-02	7.94250E-03	-1.37550E-03	2.28120E-03	2.93740E-04	5.54915E-03	4.27310E-04	4
78	5.01070E 01	5.43530E-02	2.36440E-03	-9.58040E-05	1.33570E-03	3.06310E-05	1.01876E-03	-1.09060E-04	4
79	5.04660E 01	7.59640E-02	5.77760E-03	-3.84670E-04	3.63760E-03	1.86530E-05	2.06370E-03	-1.77940E-04	3
80	5.07800E 01	3.30190E-01	4.96660E-04	-2.92760E-05	3.97130E-04	3.91140E-04	9.89662E-05	-2.82500E-05	3
81	5.12580E 01	1.88540E-01	9.58740E-03	1.83040E-04	6.31060E-03	-2.68860E-04	3.11339E-03	-2.43100E-04	4
82	5.16300E 01	7.43460E-02	2.80680E-03	-2.21310E-03	6.03990E-04	-1.27410E-04	2.18480E-03	-4.18900E-04	3
83	5.21650E 01	6.63510E-01	3.23080E-03	1.39650E-05	2.90620E-03	-4.05490E-05	3.00742E-04	3.30420E-06	3
84	5.34400E 01	1.35540E-01	2.14730E-03	5.74090E-05	1.55140E-03	3.73830E-05	5.87703E-04	-2.17280E-05	4
85	5.41190E 01	1.50210E-01	7.49660E-04	6.02010E-05	6.43290E-04	5.54510E-05	1.05371E-04	1.52650E-14	4
86	5.50640E 01	1.11170E-01	1.21740E-02	2.70600E-04	7.03310E-03	-9.58790E-05	4.87742E-03	2.12730E-04	4
87	5.58460E 01	2.51350E-01	4.60640E-03	8.84650E-04	4.20670E-03	-1.92020E-04	3.61977E-04	-1.67070E-04	4
88	5.60770E 01	1.80770E-01	1.71610E-03	-5.52340E-04	1.61100E-03	3.54360E-04	9.98644E-05	2.26220E-06	4
89	5.65040E 01	1.19920E-01	1.86810E-02	-1.14370E-03	1.18390E-02	1.14670E-04	6.22159E-03	1.55040E-04	4
90	5.78210E 01	2.21130E-01	2.83350E-03	-1.94830E-05	2.25700E-03	-1.31160E-07	5.62227E-04	-3.21640E-04	4

91	5.80580E 01	6.53540E-02	1.08280E-02	3.48570E-04	4.81140E-03	-6.08420E-06	5.74861E-03	7.83820E-04	3
92	5.86800E 01	1.36330E-01	4.82400E-03	3.63450E-05	3.39090E-03	-4.90700E-05	1.39173E-03	-1.22590E-05	4
93	5.97810E 01	2.55270E-01	6.30870E-04	-1.18680E-04	5.00910E-04	-1.19830E-04	1.29252E-04	2.69520E-05	4
94	6.01860E 01	2.55130E-01	2.35000E-03	-4.18490E-06	2.10870E-03	3.77540E-05	2.28677E-04	4.89790E-06	3
95	6.08410E 01	1.20460E-01	1.84550E-03	-6.13540E-05	1.48480E-03	2.92700E-07	3.54645E-04	-1.50980E-04	4
96	6.11250E 01	1.25360E-01	1.51780E-03	-9.27240E-07	9.19300E-04	4.71040E-05	5.93234E-04	-4.22980E-05	3
97	6.16440E 01	5.30230E-01	3.13550E-04	-3.92470E-14	3.05020E-04	-8.00520E-05	8.35522E-06	0.00000E-01	4
98	6.19000E 01	5.30170E-01	3.62710E-05	3.05710E-14	3.27070E-05	9.93580E-06	3.56166E-06	0.00000E-01	4
99	6.24000E 01	5.16240E-01	3.15360E-04	5.28000E-11	2.97690E-04	2.20530E-06	1.74932E-05	2.39240E-10	4
100	6.30200E 01	2.40090E-01	1.59860E-04	-1.88870E-12	1.18230E-04	-1.74650E-06	4.15846E-05	0.00000E-01	4
101	6.33200E 01	4.12540E-02	4.61820E-04	3.32020E-15	1.12840E-05	-8.86530E-05	4.50157E-04	0.00000E-01	4
102	6.36880E 01	6.21070E-01	8.64900E-04	-1.27290E-14	7.69230E-04	1.24870E-05	9.43401E-05	0.00000E-01	4
103	6.42930E 01	4.75450E-02	1.18580E-02	-1.49510E-08	2.04000E-03	1.98220E-04	9.49660E-03	1.76330E-06	3
104	6.57900E 01	9.64230E-02	2.05140E-03	-5.62510E-06	7.23390E-04	2.10130E-05	1.32053E-03	-7.79570E-05	4
105	6.64040E 01	8.94490E-02	2.29720E-03	-2.42060E-06	1.55890E-03	-1.96270E-05	7.28918E-04	-5.72840E-05	4
106	6.72600E 01	9.00810E-02	4.60260E-04	6.61720E-06	2.00250E-04	-4.71880E-06	2.59633E-04	0.00000E-01	4
107	6.84000E 01	2.50040E-01	1.68320E-04	2.29790E-14	1.29930E-04	-7.09060E-05	3.83396E-05	0.00000E-01	4
108	6.85300E 01	1.60110E-01	3.35730E-04	0.00000E-01	1.96030E-04	1.38810E-05	1.39500E-04	0.00000E-01	4
109	6.92930E 01	2.00720E-01	1.62230E-03	2.31690E-14	1.29010E-03	6.22940E-06	3.26184E-04	0.00000E-01	3
110	7.04040E 01	1.72720E-01	7.46520E-03	1.28180E-04	5.38060E-03	-1.05890E-05	1.98553E-03	-6.43650E-06	4
111	7.07500E 01	2.37410E-01	5.20100E-03	-7.97120E-13	3.86530E-03	1.06020E-04	1.28761E-03	-2.33120E-05	4
112	7.15700E 01	3.21020E-02	5.42150E-03	5.43740E-14	3.62650E-03	-7.08900E-05	1.72782E-03	-8.14500E-05	3
113	7.23900E 01	1.38610E-01	1.07560E-02	-4.44980E-14	6.32590E-03	3.08660E-04	4.22443E-03	-1.70830E-04	4

114	7.28750E 01	3.60370E-01	5.02370E-04	1.15400E-04	4.10810E-04	-4.30390E-05	9.09831E-05	-1.81710E-05	3
115	7.45440E 01	1.01670E-01	1.37960E-02	8.67310E-04	8.66210E-03	1.82380E-04	4.79554E-03	-4.65940E-04	4
116	7.51800E 01	2.90890E-01	1.24360E-03	1.99480E-04	9.57310E-04	-1.97210E-04	2.83541E-04	-1.79700E-06	4
117	7.55410E 01	2.33360E-01	2.65620E-03	-3.16590E-04	2.40910E-03	6.58750E-05	2.34557E-04	2.92250E-04	4
118	7.67540E 01	1.16110E-01	4.76520E-04	-1.35570E-06	3.17790E-04	1.03160E-05	1.58326E-04	2.26260E-05	4
119	7.74880E 01	1.12990E-01	4.74250E-03	-6.10330E-05	3.43590E-03	4.15440E-05	1.26662E-03	1.24340E-04	4
120	7.81200E 01	1.48220E-01	4.36310E-03	-2.75200E-05	3.02460E-03	-1.03840E-04	1.30466E-03	-2.91300E-04	4
121	7.84190E 01	1.78780E-01	6.54590E-04	2.00270E-04	4.00520E-04	5.02490E-05	2.53308E-04	-2.81380E-05	4
122	7.96760E 01	1.29790E-01	2.92290E-03	1.40950E-05	1.97520E-03	-1.21630E-06	9.28172E-04	-6.64620E-06	3
123	8.03570E 01	1.74840E-01	2.33120E-03	-5.34330E-06	1.74900E-03	5.17590E-05	5.69778E-04	-1.21120E-05	3
124	8.14350E 01	1.32040E-01	3.84110E-03	-1.69220E-05	2.57000E-03	-1.12040E-04	1.24487E-03	1.02090E-05	4
125	8.27060E 01	6.49000E-02	9.83060E-03	-2.01600E-04	2.94420E-03	-7.95650E-05	6.66551E-03	-4.86490E-04	3
126	8.35920E 01	1.18270E-01	5.35530E-03	-2.76740E-04	2.96830E-03	-6.87670E-05	2.32145E-03	-1.74030E-04	3
127	8.40600E 01	1.15000E-01	7.53230E-03	-5.00190E-04	6.46650E-03	-2.18020E-04	9.36119E-04	-2.92420E-04	3
128	8.43600E 01	1.31000E-01	7.46530E-03	-3.97290E-05	5.07020E-03	4.02030E-04	2.29602E-03	1.24970E-04	4
129	8.50100E 01	1.50000E-01	3.45460E-03	-3.42570E-05	3.00350E-03	-1.29610E-05	4.23822E-04	-7.91950E-05	3
130	8.55660E 01	1.37000E-01	1.87850E-03	1.14670E-04	1.60280E-03	6.73800E-04	2.67733E-04	1.46490E-05	3
131	8.61410E 01	1.59200E-01	3.38590E-04	-6.69290E-05	1.89990E-04	3.52700E-05	1.48396E-04	6.78080E-05	4
132	8.68740E 01	8.81200E-02	2.57860E-03	-1.84160E-04	1.67460E-03	4.40230E-05	8.88802E-04	9.71980E-06	3
133	8.75130E 01	1.90800E-01	1.69380E-03	1.14760E-04	1.41750E-03	1.07650E-04	2.71200E-04	8.19750E-05	4
134	8.84300E 01	1.78800E-01	2.53180E-03	-4.06870E-06	2.37030E-03	1.36540E-05	1.50104E-04	2.88010E-05	4
135	8.87500E 01	1.47900E-01	4.88680E-03	5.88140E-04	3.64080E-03	2.08590E-04	1.20355E-03	1.05220E-04	4
136	8.91140E 01	1.40300E-01	2.17850E-03	-4.58390E-05	1.14130E-03	-7.27970E-04	1.02876E-03	-3.81370E-04	4

137	8.98040E 01	1.30000E-01	2.58540E-03	-2.90260E-04	1.92410E-03	-7.30850E-05	6.49417E-04	1.12810E-05	4
138	9.04090E 01	5.80000E-02	3.97400E-02	-8.67100E-04	7.42220E-03	1.59060E-04	2.95102E-02	-1.86950E-03	4
139	9.12890E 01	2.80000E-01	4.88010E-03	-1.81530E-04	4.09490E-03	3.86270E-05	7.30765E-04	0.00000E-01	3
140	9.20700E 01	1.20000E-01	3.53010E-03	-1.32010E-04	1.81950E-03	-4.92470E-06	1.68845E-03	0.00000E-01	4
141	9.25900E 01	9.00000E-02	1.48890E-02	-5.19030E-05	6.34940E-03	2.53410E-04	8.14550E-03	0.00000E-01	4
142	9.32270E 01	1.20000E-01	2.02280E-03	1.64240E-04	8.44940E-04	-5.73760E-05	1.16851E-03	0.00000E-01	3
143	9.41140E 01	7.50000E-02	2.68730E-02	5.98040E-04	3.13100E-03	1.14410E-04	2.24582E-02	-6.11560E-04	4
144	9.47580E 01	1.05000E-01	3.04680E-03	6.99430E-05	1.34300E-03	2.02380E-05	1.68730E-03	1.49500E-04	4
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177	1.20300E 02	2.00000E-01	9.99330E-04	-1.41410E-05	7.19110E-04	-2.82320E-05	2.78445E-04	0.00000E-01	4
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185	1.26000E 02	1.50000E-01	1.04660E-02	6.56140E-04	7.90320E-03	3.11830E-04	2.31243E-03	0.00000E-01	3
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188	1.28190E 02	2.30000E-01	3.46570E-03	1.18220E-04	2.47230E-03	6.98610E-05	9.72047E-04	0.00000E-01	4
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195	1.33030E 02	1.12000E-01	4.14570E-03	2.93510E-04	2.71470E-03	1.41400E-03	1.40045E-03	0.00000E-01	4
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198	1.35470E 02	2.50000E-01	7.73160E-03	4.66010E-05	9.78100E-03	2.41500E-04	8.13965E-04	0.00000E-01	3
199	1.36360E 02	9.00000E-02	1.72000E-02	-1.42190E-04	5.84260E-03	1.77190E-04	1.08315E-02	0.00000E-01	4
200	1.37530E 02	5.90000E-02	2.82700E-02	2.86750E-03	1.05040E-02	-1.77920E-04	1.65452E-02	0.00000E-01	4
201	1.39190E 02	4.50000E-02	6.42530E-03	1.02700E-03	2.92230E-03	-1.95030E-06	3.42961E-03	0.00000E-01	4
202	1.40240E 02	1.06200E-01	4.51470E-03	6.38890E-04	2.74740E-03	1.26440E-05	1.73106E-03	0.00000E-01	4
203	1.41810E 02	1.00000E-01	7.11520E-03	4.32670E-04	4.45390E-03	-3.90790E-06	2.57130E-03	0.00000E-01	4
204	1.42140E 02	9.50000E-02	2.44450E-02	1.44340E-03	1.04020E-02	1.09380E-03	1.29807E-02	0.00000E-01	4
205	1.43050E 02	9.33000E-02	1.22800E-03	2.79760E-04	1.08100E-03	1.37830E-04	1.44319E-04	0.00000E-01	4
206	1.45600E 02	1.05100E-01	2.54190E-02	-1.27870E-03	1.23500E-02	-1.67130E-03	1.19203E-02	0.00000E-01	4
207	1.46350E 02	1.20000E-01	2.23950E-03	0.00000E-01	1.04520E-03	-8.69160E-04	1.18536E-03	0.00000E-01	4
208	1.47350E 02	7.30000E-02	1.83250E-02	0.00000E-01	9.21680E-03	1.12300E-03	8.51121E-03	0.00000E-01	4

EVALUATION OF NUCLEAR DATA FOR ^{235}U IN THE UNRESOLVED RESONANCE REGION

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ABSTRACT

The mean resonance parameters of ^{235}U in the 0.1-100 keV region were obtained using the data both on the resolved resonances and on the cross-sections in the keV region, taking their structure into account. With the mean parameters obtained, the quantity α can be described satisfactorily for the entire region. Variation of the quantity $\langle\Gamma_\gamma\rangle$ is necessary to obtain full agreement with the estimated values of α.

The unresolved resonance of ^{235}U ranges from 0.1 to 100 keV. We can restrict our discussion here to the contribution of the S- and P- waves. The existence of five excitation levels in this region makes it necessary to take the inelastic neutron scattering reaction into account. The mean resonance parameters of ^{235}U in the 0.1-100 keV region cannot be obtained from the resonance region data alone, as that region includes only S-states, which are moreover difficult to identify by their spin. Besides, the ^{235}U nucleus has an intermediate structure in the σ_f and σ_γ cross-sections in the 10-40 keV region [1] which is not reflected by the available data on resolved resonances. Thus, some of the mean parameters for this paper were obtained from resolved resonance data, others by fitting to the data on the cross-sections in the keV region.

The average distance between the levels $\langle D \rangle_J$ was determined on the basis of an independent particle model [2], assuming independence of neutron energy. The fundamental theoretical parameter for level density α was obtained from the average observed distance between the levels $\langle D \rangle$ exp. in the resolved resonance region [3], equal to 0.61 eV. The value of α was found to be 28.61 MeV^{-1}.

The mean neutron widths $\langle \Gamma_n \rangle_S$ of the S-states, which are determined by the values of ℓ and J, were obtained from the strength functions S_o and S_1; these were estimated for this paper by fitting the calculated data on σ_t to the experimental results in the entire energy region. The potential scattering cross-section in the low energy region was taken to be 11.5 barn [4]. In the calculations, the absence of inter-resonance interference was assumed. The following formula was used:

$$\langle \sigma_t \rangle = \sum_\ell 4\pi(2\ell+1)\frac{\sin^2\varphi_\ell}{K^2} + \sum_\ell (2\ell+1)\frac{2\pi^2}{K^2} S_\ell P_\ell - \qquad /1/$$
$$-\sum_\ell 2(2\ell+1)\frac{2\pi^2}{K^2} E^{1/2} S_\ell P_\ell \sin^2\varphi_\ell ,$$

where φ_ℓ is the phase shift and P_e the penetration factor for a given partial wave. The results obtained were:

$S_o = 1.08 \times 10^{-4} eV^{-1/2}$, $S_1 = 1.58 \times 10^{-4} eV^{-1/2}$, which agrees with the data of other authors. Concerning S_o, we should note the good agreement with the result of an independent evaluation in the unresolved resonance region [3], which gives the value $S_o = 1.069 \pm 0.14 \times 10^{-4} eV^{-1/2}$.

A comparison of the calculated and evaluated data on σ_t is given in Fig. 1.

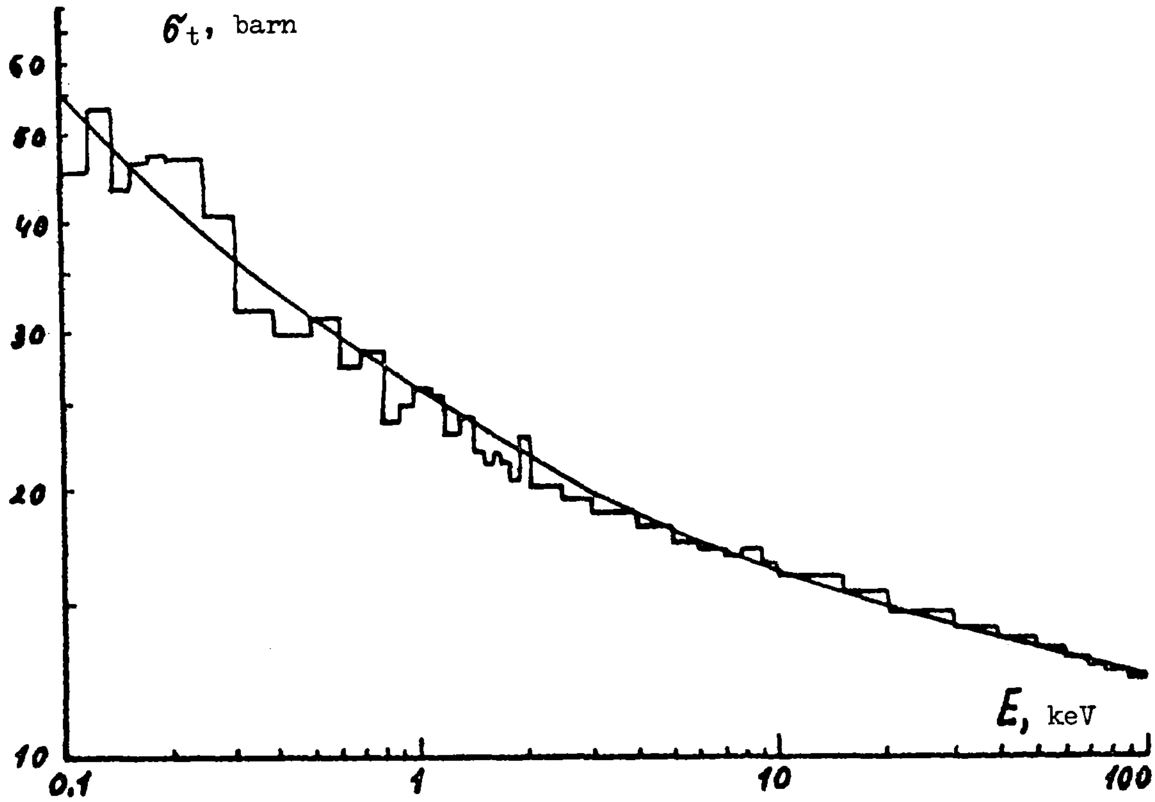

Fig. 1 Comparison of calculated and evaluated data on σ_t.

As subsequent calculation was based on formulae for average cross-sections in sufficiently narrow energy ranges, the structure of the cross-section σ_t was taken into account by the change in S_0. This was found to be necessary for obtaining agreement of the calculated and experimental data on α (cf. Ref. [5]).

The mean radiation width $\langle \Gamma_\gamma \rangle$ was determined from the radiation widths of the resolved resonances and was taken to be 0.0407 ev[3]. Variation of the quantity $\langle \Gamma_\gamma \rangle$ from this value does not lead to improvement of agreement on the quantity α in the entire energy region.

The mean inelastic widths $\langle \Gamma_{n'} \rangle_S$ were obtained in the same way as the neutron widths $\langle \Gamma_n \rangle_S$, taking into account the possibility of neutron escape through various outlet channels (q, ℓ'), where q indicates the excited level with energy E_q and ℓ' the orbital momentum of the escaping neutron:

$$\langle \Gamma_{n'} \rangle_s = \langle D \rangle_J \sum_{q,\ell'} f_{\ell'} \mathcal{E}_q^{1/2} P_{\ell'}(\mathcal{E}_q)\, V_{J,\ell',q}\,, \qquad /2/$$

where $\mathcal{E}_q$ is the neutron's energy in the outlet channel, and $\nu_{J,\ell',q}$ is the number of its degrees of freedom. The (n,n') reaction in the unresolved resonance region must be taken into account in calculating the cross-sections σ_f and σ_γ. In Fig. 2, calculations of σ_f with and without allowance for competition from inelastic scattering are compared. It will be seen that the effect amounts to 10% at 100 keV.

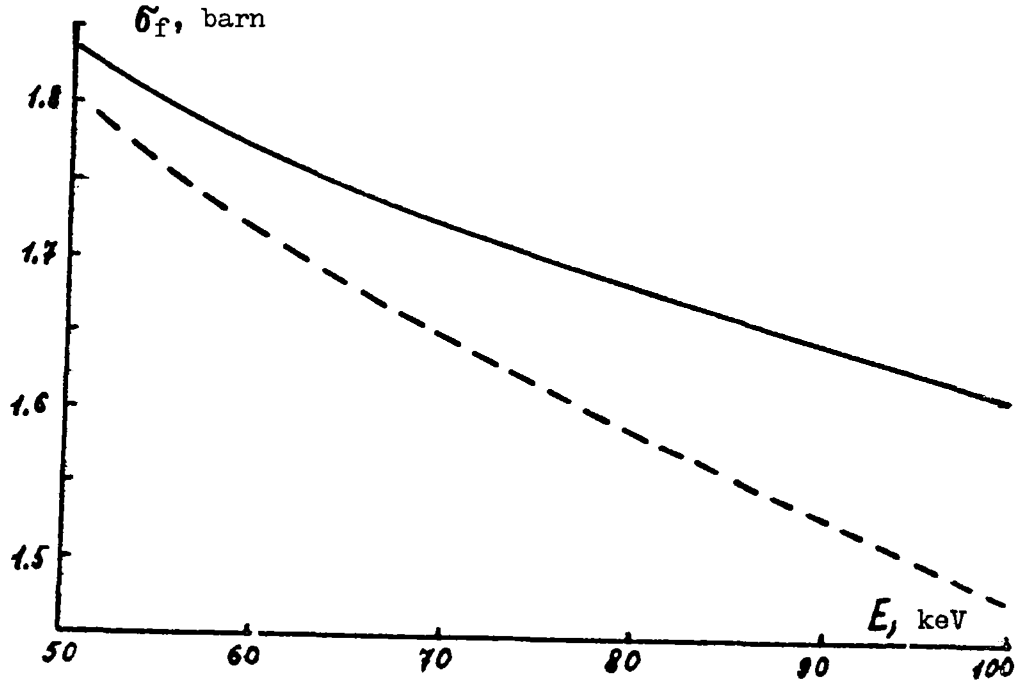

Fig. 2 Comparison of cross-sections σ_f, calculated with (- - -) and without (——) allowance for competition of the (n,n') reaction.

The mean fission widths $\langle \Gamma_f \rangle_S$ were calculated on the basis of the channel theory of fission:

$$\langle \Gamma_f \rangle_J = \sum_K \frac{\langle D \rangle_J}{2\pi} P(E_{fK}, \hbar\omega_K), \qquad /3/$$

where $P(E_{fk}, \hbar\omega_k)$ is the penetration factor of the k^{th} fission barrier, characterized by the height E_{fk} and the curvature parameter $\hbar\omega_k$. The penetration factor was calculated using the well-known Hill-Wheeler expression. The same value of the parameter $\hbar\omega_k$, 0.5 MeV, was taken for all channels. The barrier heights E_{fk} were obtained by fitting calculated data on σ_f to the experimental results. In doing this, we bore in mind the model system of transition states of the even-evenly splitting nuclei put forward by Lynn [6]. A comparison for σ_f is given in Fig. 3.

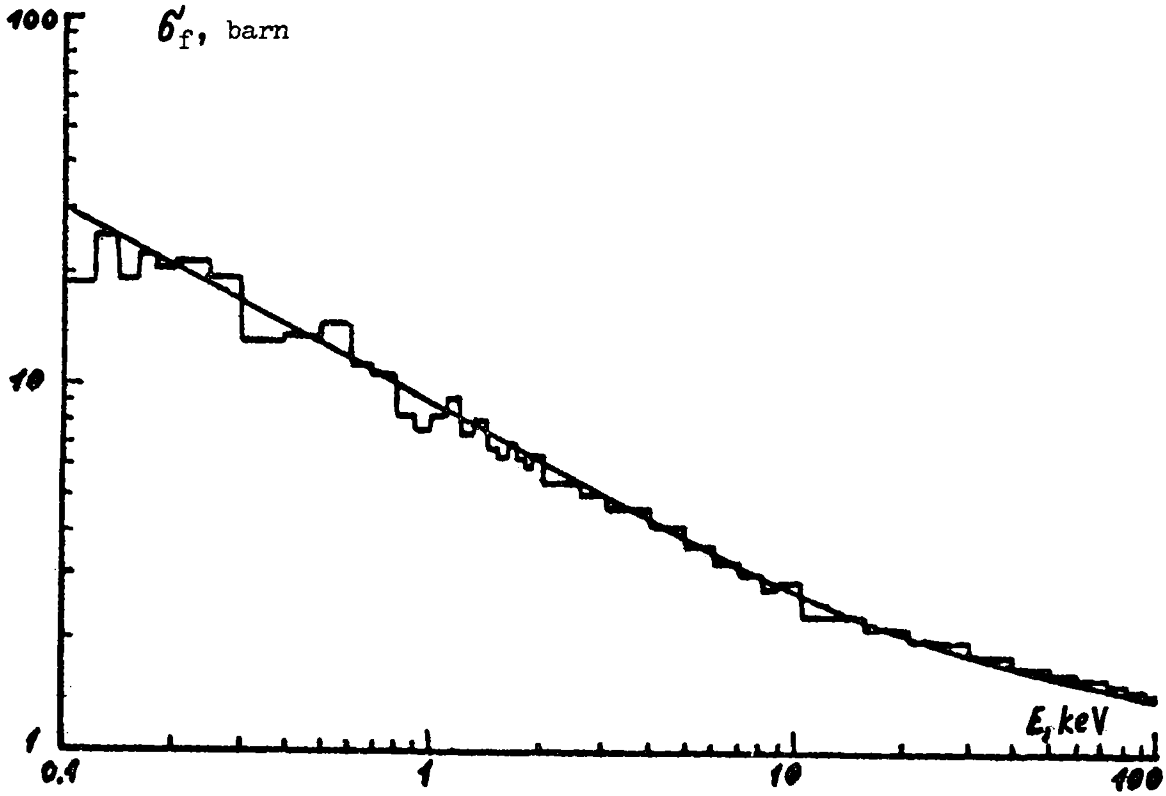

Fig. 3 Comparison of calculated and evaluated data on σ_f.

Subsequently, the partial width $\langle\Gamma_f\rangle_4^0$ for l = 0 and J = 4 was fitted to σ_f for each energy range in order to take account of fluctuations in σ_f.

Obtaining mean resonance parameters taking into account the structure of the σ_t and σ_f leads, in calculation, to a corresponding strong structure in σ_γ (Fig. 4) and α.

The criterion for the quality of the results obtained is comparison of the calculated and experimental data on α. As was shown in Ref. [5], the agreement is fully satisfactory. To improve the agreement with the evaluated data on α obtained by analysis of the experimental data provided in Ref. [5], one must vary the quantity $\langle\Gamma_\gamma\rangle$ in each energy range.

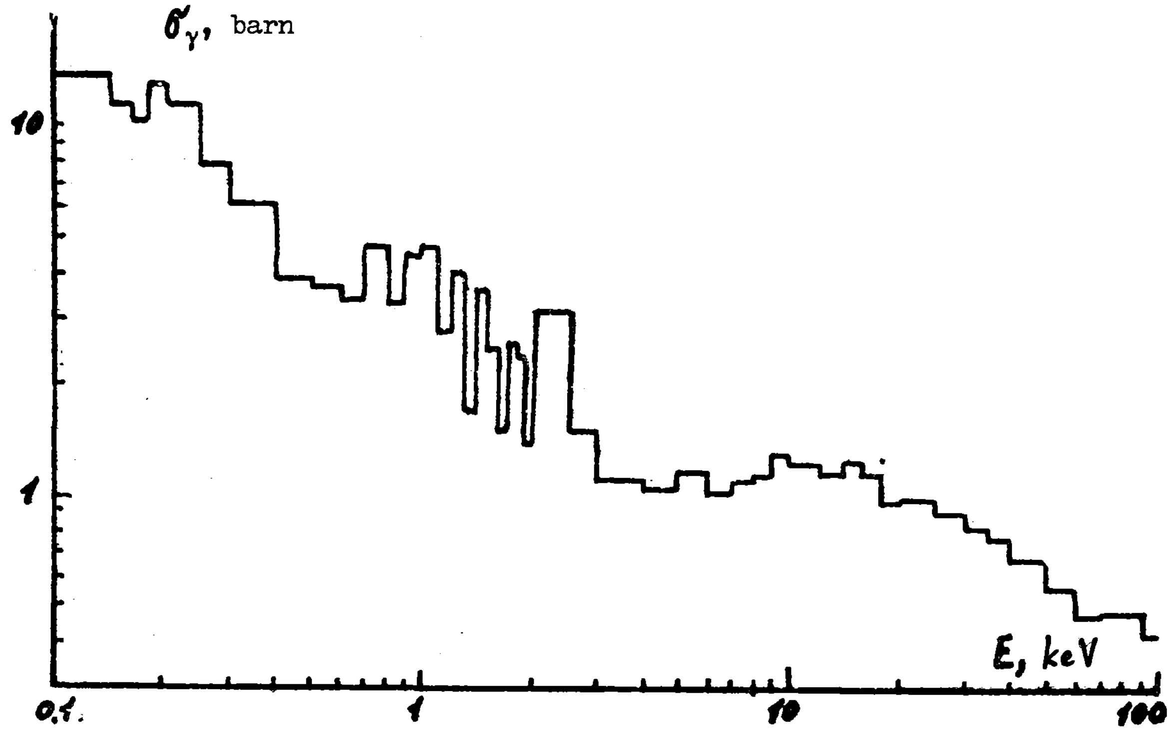

Fig. 4 Calculated cross-section of neutron radiation capture, σ_γ.

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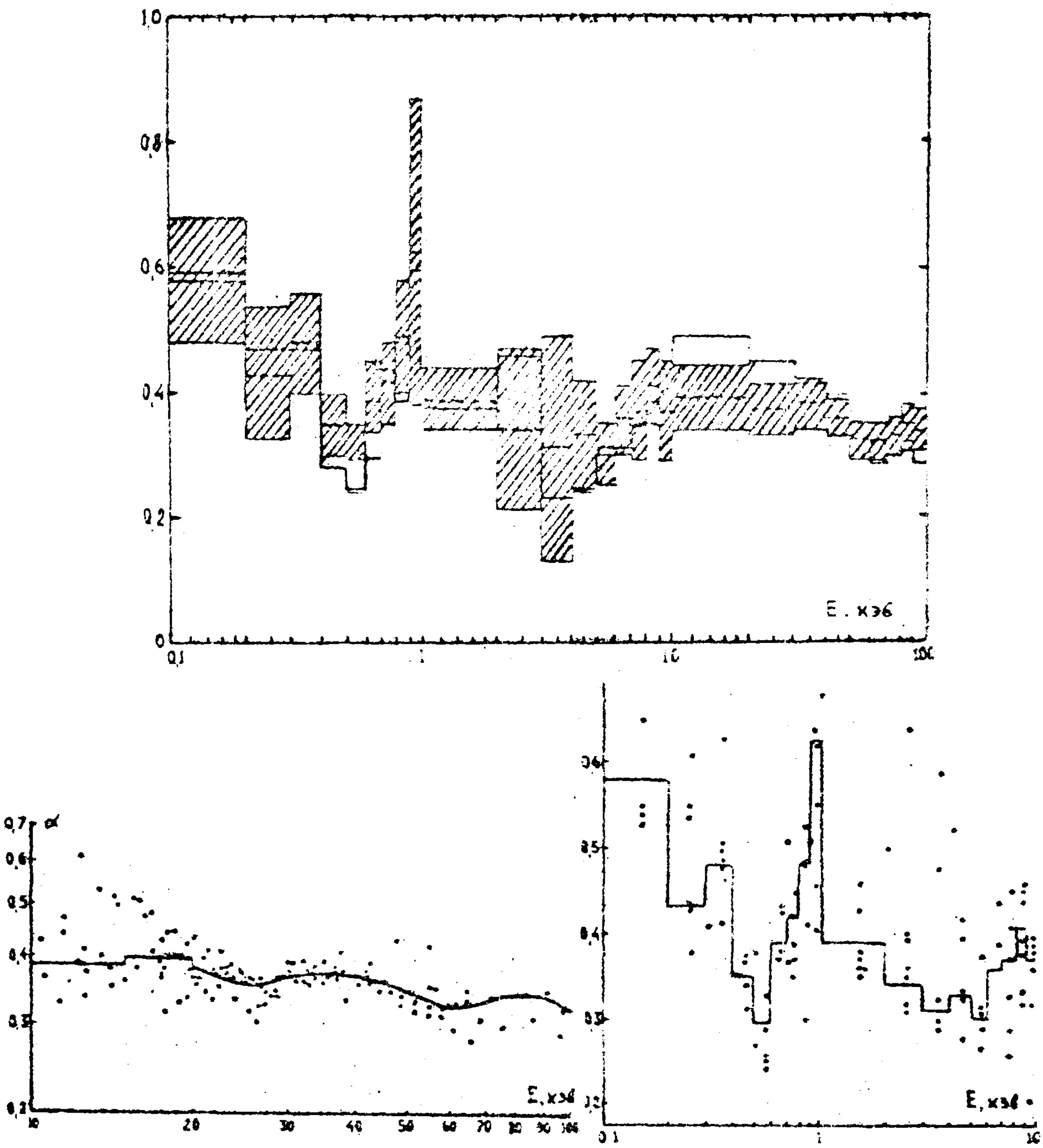

Fig. 1 a) comparison of calculated (———) and evaluated (- - - -) data on α.

evaluated error corridor

b) experimental and evaluated data on α in the 10–100 keV energy region.

c) experimental and evaluated data on α in the 0.1–10 keV region.

EVALUATION OF α (^{235}U) IN THE ENERGY RANGE 0.1 keV-15 MeV

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ABSTRACT

The available experimental data on α (^{235}U) in the energy region above 100 eV were analysed with a view to finding possible systematic errors in experiment. The value of α was calculated on the basis of the mean statistical parameters evaluated by us. The influence of fluctuations of the force function S_0 and changes in $\langle\Gamma_\gamma\rangle$ on the results of the calculations of α was examined.

During the last 15 years, the quantity α (^{235}U) has frequently been measured, both directly (18 studies) and indirectly (4 studies - measurement of η and σ_α). The main shortcoming of these studies is that α was measured in a limited energy region, and normalization to the thermal energy region of neutrons was almost never used.

The α (^{235}U) was evaluated in two overlapping regions: from 100 eV to 30 keV and from 20 keV to 1 MeV. The measurement results for α (^{235}U) in the energy region below 20 keV are given in Figs. 1 and 2. These figures show that the results of available measurements in the standard energy ranges are not in good agreement with each other, differing in some cases by a factor of 1.5.

The reasons why the experiments do not agree are that they were not all normalized in the same way, that the errors were underestimated in some experiments, and that there are errors in the experimental measurement methods.

The simplest way of checking the normalization of α is to compare the α-values obtained or used for well-resolved resonances of ^{235}U. Unfortunately, hardly any authors give α-values in the resonance region in which normalization was performed. In some cases, authors normalized for the integral of fission and capture in different regions. Analysis of the normalization in Refs. [1-6] showed that it is apparently unnecessary to change it i.e. one can analyse all experiments after normalization by the same method.

It is difficult to evaluate how real the errors cited by the authors are. In some energy ranges the spread of the data is greater than the experimental errors given by the authors. Measurement of α consists of measuring the number of fissions N_f and the number of captures N_γ. The effect-to-background ratio is higher for N_f, which means that the background uncertainties in N_γ lead to greater errors in α than the background uncertainties in N_f. From measurements of N_f, the quantities σ_f can be obtained, and since the background is low, the results of different experiments should agree with each other. If an experiment runs counter to the general tendency of σ_f, this would suggest that there may be errors in the background measurement, which would probably also affect the measurements of N_f. However, such a comparison of σ_f(^{235}U) does not give the desired result, as the authors of only three papers (Refs. [1,6,5]) give values of σ_f that agree well with the results of other authors. Moreover, the results of some experiments, e.g. the one described by Kurov et al. [4], are very insensitive to the "σ_f criterion", but very sensitive to scattered neutrons. We believe that there is no reason for assigning a lower weight to the experimental data in question on the basis of the σ_f criterion.

Comparison of experimental methods of measuring α (^{235}U) indicates, above all, that their sensitivity varies (number of recorded constants). The most sensitive methods are those used by Muradyan et al. [3], Kurov et al. [4], Van-Shi-Di [5]; less sensitive are those used by De Saussure et al. [1] and Perez et al. [6] and the least sensitive methods are those used by Czirr and Lindsey [2], Bandl et al. [7] and Vorotnikov et al. [8]. Analysis of possible systematic errors in different experiments is meaningful for four characteristics: operation of the gamma-ray detector, operation of the fission detector, background determination, energy resolution.

Gamma-ray detectors must be insensitive to changes in the γ-ray spectrum from capture or fission, and to the full energy of fission γ-rays. In Czirr and Lindsey's experiment a modified Moxon-Rae detector with a very small fission and capture effectiveness ratio $E_f/E_\gamma = 0.86$ (expected value: 1.0-1.3) was used. The Moxon-Rae detectors in use at present show a spread of 0.8 to 1.5 in the E_f/E_γ ratio. As it is not known which value is correct, we reduced the weight of Czirr and Lindsey's experimental values (adding a squared error of 5%). The liquid scintillators used in the work described in Refs. [1,5, 6,4] are in principle more sensitive to changes in the capture γ-ray spectra than the Moxon-Rae detectors. In the experiment described by Kurov et al. [4] use was made of the agreement between the two halves of the scintillator, which may have caused errors. In the experiments of Muradyan et al. [3] and Vorotnikov et al. [8] a certain sensitivity to changes in the capture and fission γ-ray spectre is also possible.

The methods applied to record fission events are imperfect with respect to possible sensitivity to changes in the characteristics of fission processes depending on the energy of incident neutrons. However, the errors arising from this are apparently negligible at energies below 10 keV. In principle, there may be an additional error in experiments where the α depends on $\bar{\nu}$, if $\bar{\nu}$ changes according to the spin of the compound nucleus. This applies to the experiments outlined in Refs.[2,4,5]. An additional 5% uncertainty was introduced for this effect.

The most serious error in measuring α is connected with background determination. If the background is measured with filters resonance, the results of measurements at an energy higher than that of the filter will, of course, be unreliable, and should be given less weight. Therefore the results of Czirr and Lindsey's measurments [2] in the region above 3 keV should be given less weight (the background was not measured at an energy higher than 2.8 keV).

In the experiment by Muradyan et al., background measurements were hampered, especially in the region above 900 eV, and the Nγ count was rather low, so their results were given less weight (the error up to an energy of 1 keV was increased by 10%). In the experiments described in Refs. /4 and 5/ there is a considerable sensitivity to scattered neutrons, so the error in α was increased by 20%.

Errors can arise in experiments if delayed fission γ-rays are recorded as capture events. At energies below 30 keV these γ-rays can introduce into α an error of the order of $\pm$ 0.02 or less. We took this systematic error into account in all our experiments.

As α has structure, energy resolution is important. Apparently, the minimum number of resolution widths fitting into the averaging ranges must be 2 (in which case ~ 12% of reactions are caused by neutrons of different energies). Therefore, we must give less weight to the Czirr et al. measurements /2/ in the region above 5 keV, to those of Kurov et al. /4/, Van-Shi-Di et al. /5/ and Bandl et al. /7/ in the region above 8 keV, and to those of Vorotnikov et al. /8/ in the region above 10 keV.

In evaluating α, a 5% squared error was added to the authors' errors for each observation made above (except for errors connected with background determination). The total error in the evaluated values of α (^{235}U) is 10-20%, consisting of the systematic error (7%), determined by normalization of α (~5%), delayed γ-rays (~5%) and some other factors (~ 3%) and of the random error determined from the spread of the experimental data, taking into account the "weights" of the experimental points.

In the energy region above 20 keV the four main series of α that are absolute - the data in Refs. /9-12/ - were obtained essentially by the same method, and as a result identical systematic errors were possible. In order to normalize these data uniformly an energy range common to all these studies was chosen, namely, 30 $\pm$ 10 keV. The mean value of α in this range was obtained using the α values given in Refs. /9,10,11,13/. As the results of the experiments described in Refs. /12 and 13/ do not agree with each other, the results of the earlier work /12/ were considered unreliable and were not used to obtain the mean value of α at 30$\pm$10 keV.

Apart from the evaluation of experimental data on α, this quantity was calculated using the mean resonance parameters given by Antsipov et al. /14/. A comparison of calculated and experimental data is given in Fig. 3. As will be seen, agreement is satisfactory. The calculated structure of α agrees with the experimental results. It should be noted that the calculation results are estimated somewhat higher in the 10-30 keV region. Unfortunately, the low accuracy of the experimental data does not permit evaluation of the reliability of the calculated values of α in this region.

As the calculations were made in narrow energy groups with expressions for the mean cross-sections, the problem of the effect of fluctuations in the strength function S_0 (or alternatively in the neutron widths of the S-wave) corresponding to fluctuations of the total cross-section σ_t on the results of calculation of α was studied. The calculation showed that ignoring this structure in the region below 10 keV leads to loss of agreement with the experiments for the selected width of the groups (up to 1 keV).

Apart from this, the problem of the effect of the value of the mean radiation width $\langle \Gamma_\gamma \rangle$ on the results of calculation of α was examined. This was due to the fact that the value of $\langle \Gamma_\gamma \rangle$ can differ by factor of almost 2 in different evaluations. The calculation showed that variation of $\langle \Gamma_\gamma \rangle$ from the selected value 0.0407 eV does not improve agreement with experiment throughout the region. With an increase of $\langle \Gamma_\gamma \rangle$ from 0.025 to 0.045 eV, i.e. by a factor of 1.8, the value of α increases 1.05 - 1.5 times. This effect increases with lower values of α and higher energies.

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ANALYSIS OF THE ANGULAR DISTRIBUTIONS OF ELASTICALLY SCATTERED NEUTRONS FOR ^{235}U

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ABSTRACT

Experimental data on the angular distributions of 0.5-15 MeV neutrons elastically scattered by ^{235}U nuclei are analysed on the basis of Bessel functions and Legendre polynomial expansions. The advantages of the method are that there are no negative cross-sections and relatively few expansion coefficients and that experimental data on scattering at 0^{o} and 180^{o} are not needed.

A knowledge of the angular distributions of elastically scattered neutrons is required for accurate prediction of the behaviour of neutrons passing through matter and for obtaining more precise parameters for the optical model of the nucleus. The standard method for analysing the angular distributions of elastically scattered neutrons is Legendre polynomial expansion. The degree of expansion of the scattering amplitude is equal to the highest momentum of the neutron undergoing scattering, i.e. approximately 30 expansion terms are necessary for an incident neutron energy of the order of 14 MeV. Measurements are usually made at 15-20 different angles, i.e. the set of Legendre polynomial expansion coefficients carries more information than is contained in the experimental data.

The physical inaccuracy of such a description is well known. At a sufficiently high degree of Legendre polynomial expansion the curve describes the experimental points, but in the spaces between them it can behave quite unphysically, giving negative cross-section values. In principle, moreover, experiments on scattering do not permit measurement of the differential cross-sections at extremely small and large angles, and for this reason the fitting procedure must permit extrapolation to these angles. Precisely because of the orthogonality of the Legendre polynomials this is impossible, although scattering at small angles constitutes a large portion of the total cross-section at energies above 8 MeV.

The Bessel functions have certain advantages for describing angular distributions. Such a description requires a smaller number of expansion terms, and use of the Bessel functions enables one to trace the dependence of the angular distributions on neutron energy, nucleus dimensions, scattering angles, and also to obtain the values of the scattering cross-sections at angles of 0^{o} and 180^{o}.

It is well known that in the approximation of small angles and for an absolutely black scatterer the series of the scattering theory is summed accurately (see Ref. [1]), and for the differential scattering cross-section we have

$$\frac{d\sigma}{d\Omega} = (KR^2)^2 \left[\frac{J_1(x)}{x} \right]^2 , \qquad (1)$$

where $J_1(x)$ is the Bessel functions, $K = 2\pi/\lambda$, $x = 2KR\sin\theta/2$, R is the radius of the nucleus and θ is the scattering angle. Even in such simple form the formula (1) correctly describes the value of the leading cross-section peak and the location of the second.

The experimental data on elastic scattering always contain a component from the inelastic scattering at low levels. Besides, taking into account the diffuseness of the nucleus boundaries leads to a certain complication of formula (1) [2]. A simple theory [3] gives the following expressions for the differential inelastic scattering cross-sections associated with quadrupole and octupole oscillations of the nucleus:

$$\left(\frac{d\sigma}{d\Omega}\right)_{\text{квадр.}} = (KR^2)^2 \frac{5}{8\pi} \frac{E_2}{C_2} \left[\frac{1}{4} J_0^2(x) + \frac{3}{4} J_2^2(x)\right] \qquad (2)$$

$$\left(\frac{d\sigma}{d\Omega}\right)_{\text{октуп.}} = (KR^2)^2 \frac{7}{8\pi} \frac{E_3}{C_3} \left[\frac{3}{8} J_1^2(x) + \frac{5}{8} J_3^2(x)\right], \qquad (3)$$

where E_i is the excitation energy and C_i is the surface tension energy. We shall therefore describe the angular distributions of the elastically scattered neutrons by the following formula:

$$\frac{d\sigma}{d\Omega} = (KR^2)^2 \left\{ \mathcal{D}\left[\frac{J_1(x)}{x}\right]^2 + \sum_{i=0}^{M-1} A_i J_i^2(x) \right\}, \qquad (4)$$

where M is the number of Bessel functions and D and A_i are the fitting parameters. Integrating (4) we obtain the integral scattering cross-section

$$\sigma_s = \pi R^2 \{ \mathcal{D}[1 - J_0^2(2KR) - J_1^2(2KR)] +$$
$$+ (2KR)^2 \sum_{m=0}^{M-1} A_m [J_m^2(2KR) + J_{m-1}^2(2KR) - \frac{2m}{2KR} J_m(2KR) J_{m-1}(2KR)]\} \qquad (5)$$

We have written a program enabling us to perform fitting to the experimental data in formula (4). It was found that in expansion of the angular distributions by Bessel functions a significantly smaller number of terms from series (4) is necessary than if Legendre polynomial expansion is used. Moreover, no previous knowledge of the differential scattering cross-sections at the angles 0° and 180° is needed. The scattering cross-section values at these angles obtained by Bessel function fitting were applied by us to obtain the Legendre polynomial expansion. The Bessel function expansion has yet another important advantage. It explicitly embodies energy dependence, thus permitting interpolation into energy regions for which there is no experimental information.

The approach outlined above was used to analyse the experimental data on the angular distributions of elastically scattered neutrons for ^{235}U. The following six sets of experimental data in this region are available: Allen et al. [4], Batchelor and Wild [5], Knitter et al. [6], Cranberg [7], Smith and Guenther [8] and Kammerdiener and Luther [9]. Smooth curves were drawn through the experimental points on the basis of Bessel function expansion. The integral scattering cross-sections obtained by integrating these smooth curves were regarded as experimental values and used to evaluate the integral cross-section of elastic neutron scattering. The standard practice is to represent differential elastic scattering cross-sections as Legendre polynomial expansions. To obtain such an expansion we used, instead of the experimental values of the differential cross-sections, smoothly interpolated cross-sections obtained from Bessel function expansion, considering them "true" and, in keeping with this, assigning the same relative weight to them. In each case 101 points were used, uniformly distributed along cos θ over the range -1 to 1. Figure 1 shows the characteristic situation that arises when the experimental data are described by differential cross-sections using either Bessel functions or Legendre polynomials.

The elastic neutron scattering distributions obtained are sufficiently reliable in the energy region up to 6 MeV, but in the 6-14 MeV energy range, where experimental data are entirely lacking, the distributions were obtained by using the energy dependence of the Bessel function expansion and interpolating between the points 5.5 MeV and 14 MeV.

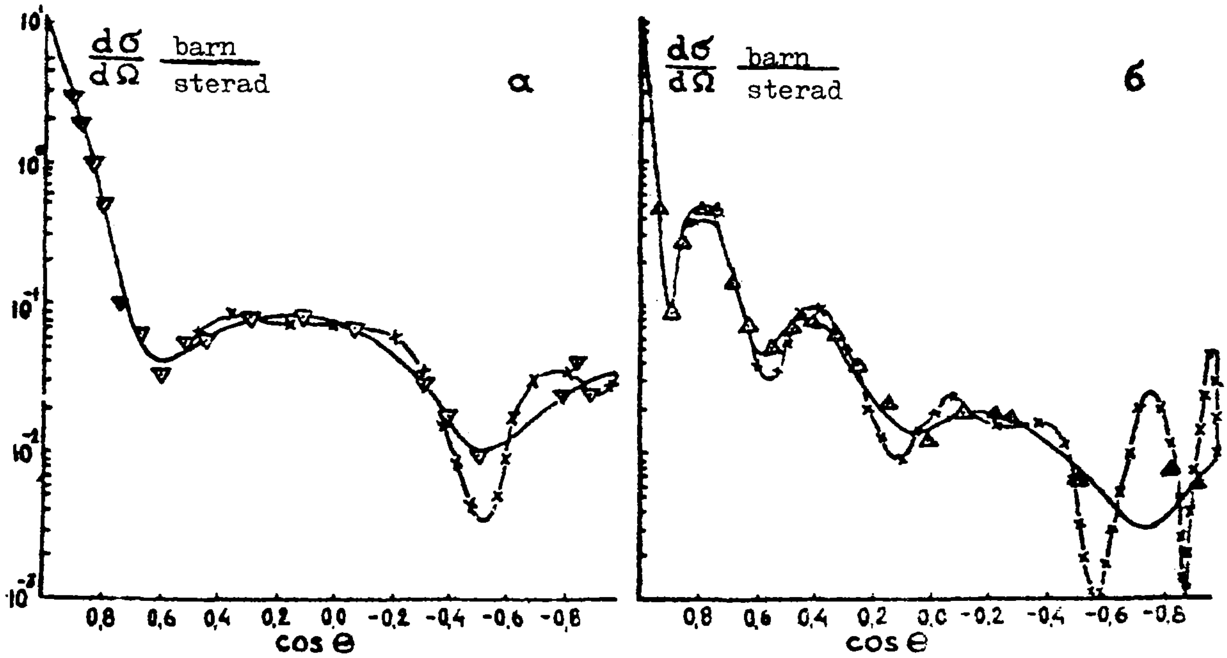

Fig. 1 Angular distributions of elastically scattered neutrons

a) Measurements by Knitter [6] at an energy of 4.5 MeV;

b) Measurements by Kammerdiener [9] at an energy of 14 MeV.

——x——x—— Legendre polynomial expansion not using additional information;

———————— Bessel function expansion.

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EVALUATION OF THE INELASTIC NEUTRON SCATTERING CROSS-SECTION FOR ^{235}U

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ABSTRACT

The inelastic neutron scattering cross-section for ^{235}U was calculated in the region of resolved and overlapping levels of a target nucleus (0-3.5 MeV) on the basis of a statistical model taking into account competition from capture and fission. On the basis of these theoretical calculations, and of an analysis of the experimental information on $\sigma_{nn'}$ evaluated data on $\sigma_{nn'}$ (^{235}U) in the energy region up to 15 MeV were obtained.

Only an insignificant amount of experimental information on the cross-section of inelastic neutron scattering by the ^{235}U nucleus is available. Theoretical calculations are also difficult because of the competition from fission and capture and the deformity of the ^{235}U nucleus. The calculations on ^{235}U known to us (Refs. [1-3]) are more than ten years old, and in the light of modern knowledge of the level structure of the ^{235}U nucleus there are clearly not enough of these calculations.

The ^{235}U level scheme has been investigated fully in recent years. The experimental data of the various authors agree, or complement each other, in the region up to E $\sim$400 keV (see the papers by Cline [4], Stephens [5], Braid [6,7], Rickey et al. [8]). We have limited our study to the region up to 725 keV, as the number of levels not identified in Ref. [8] increases sharply above this region. Below 725 keV only four levels are unidentified. The region up to 725 keV contains 50 levels, which is quite sufficient for calculating the inelastic neutron scattering cross-section in the region of resolved levels of the ^{235}U target nucleus. In the region above 725 keV it is advisable to use the continuous level spectrum approximation as the number of levels becomes very large.

The level scheme of the ^{235}U nucleus used by us differs somewhat from that recommended by Schmidt [9], and is more extensive. One should, however, note that Rickey et al. [8] consider only the characteristics of the first five bands as firmly established, but in the region of higher target nucleus excitations it is entirely feasible to change the proposed level system.

We used the evaluated scheme to calculate the inelastic neutron scattering cross-section and the level excitation cross-sections of the ^{235}U nucleus in the energy range up to 3.5 MeV on the basis of a statistical model. The calculations were made with our NERIS programme, in which a Hauser-Feschbach formalism modified by including the effect of neutron and fission width fluctuations is developed.

The inelastic neutron scattering cross-section with excitation of a level of energy $E_q < E_{q_{max}}$ has the form:

$$\sigma_{nn'}^{E_q}(E_n) = \frac{\pi}{k^2}\frac{1}{2(2I+1)}\sum_{l,j} T_{lj}(E_n)\sum_J (2J+1)\frac{\sum_{l'j'} T_{l'j'}(E_n - E_q)\, R^{J\pi}_{\gamma\alpha'}}{\sum_\gamma T_\gamma(J,E_n) + T_f(J\pi,E_n) + \sum_{l'j'p'} T_{l'j'}(E_n - E_{p'})}$$

where K is the neutron wave number, i is the spin of the ground state of the target nucleus, l and j are the orbital and total momentum of the incident and escaping (l', j') neutron and J is the spin of the compound nucleus. The indices with two primes refer to all the neutron channels satisfying the laws of conservation of energy, parity and momentum. The first term of the denominater, which takes competition from radiation capture into account, can be written as follows on the assumption that Γ_γ is independent of energy and spin:

$$\sum_{l'} T_{l'}(J,E_n) = 2\pi \frac{\Gamma_\gamma}{D(J,E_n)}$$

The effective fission penetration factor $T_f(J,\Pi,E_n)$ was calculated on the basis of the channel theory of fission. The main problem is to select fission barrier parameters that will allow the fission cross-section in a given energy region to be described sufficiently well. We achieved this up to an energy of 100 keV. On the basis of this work and also of papers by Garrison [10] and Otter [11] we selected the fission barrier parameters and the values of the degrees of freedom V_t for the distribution of the fission widths, which were associated with the number of open fission channels.

In order to calculate the neutron penetration factors in terms of an optical model, the programme used the local potential $V(\gamma)$, which contained a real part, an imaginary term describing the surface absorption, and real spin- an orbital term.

By introducing a correction factor $R^{J\Pi}_{\alpha\alpha'} = \left\langle \frac{\Gamma_\alpha \Gamma_{\alpha'}}{\Gamma} \right\rangle \Big/ \frac{\langle\Gamma_\alpha\rangle\langle\Gamma_{\alpha'}\rangle}{\langle\Gamma\rangle}$ one can correct the excessively powerful assumption as to the dependence of decay of the compound nucleus on how it was formed.

In the present work we limited our examination of level excitation cross-sections to the region up to $E_{q\ max} = 414.76$ keV. Above this region the levels are so dense that the excitation spectrum can be considered continuous. Nor did we pay attention to levels with $l_0 > \frac{11}{2}$ in our calculations, as their contribution to $\sigma_{nn'}$ is small because of the great difference in spin from the ground state. In our calculations, we used the parameters of a spherical optical potential (see Ref. [21] *), which satisfactorily describes the energy dependence $V\sigma_t$ and $\sigma_{el}R$ up to 3 MeV: $V_o = 45$ MeV, $W_o = 10.7$ MeV, $V_{so} = 10$ MeV, $R_v = R_w = 1.25$ fermi, a = 0.67 fermi, b = 0.98 fermi.

* Translator's note - only 20 references are listed.

As initial information for obtaining evaluated data on $\sigma_{nn'}$ (^{235}U), we used, first, the results of calculation by the method described above; second, more or less direct experimental data on $\sigma_{nn'}$(^{235}U): the measurements by Armitage et al. [12] for 130, 400, 550, 710, 1000 and 1500 keV; Knitter et al. [13] at neutron energies of 1.5, 1.9 and 2.3 MeV; Batchelor and Wyld [19] at 2, 3 and 4 MeV; Cranberg [15] at 0.55, 0.98 and 2.0 MeV; Drake [16] at 4.0, 6.0 and 7.5 MeV; Smith [17] at 0.517 MeV; Allen [18] at 0.25, 0.5 and 1.0 MeV and Beyster et al. [19] at 1.0 and 2.5 MeV; and thirdly, data on $\sigma_{nn'}$ obtained from the difference of the cross-sections.

In the region of resolved target-nucleus levels mainly theoretical calculations were used for the evaluation. In the region of higher energies, where the excitation cross-sections of the separate levels are still sufficiently high (up to 4 MeV), data on $\sigma_{nn'}$ obtained from the difference of the cross-sections were used in addition to the results of the theoretical calculations. Thus, the level excitation cross-sections were taken from the calculations, while the inelastic scattering cross-section component responsible for the excitation of the continuous level spectrum σ_{cont} was calculated in the form of the difference

$$\sigma_{cont} = \sigma_{n'} - \sum \sigma_{n'}^{E_q}$$

where $\sigma_{n'}$ was obtained from the difference of the evaluated cross-sections

$$\sigma_{n'} = \sigma_{nx} - \sigma_{f} - \sigma_{\gamma}$$

And finally, the cross-section in the energy region above 3.5 MeV was obtained from the difference of the cross-sections.

The evaluated values of $\sigma_{nn'}$ (^{235}U) in the region up to 2 MeV and the data for the separate groups of levels are given in Fig. 1.

In the energy region up to 2 MeV, the theoretical calculations agree satisfactorily with the experimental data. The data evaluated in the present work are higher than those of the English data library in this region. The evaluated data differ particularly in the region near the threshold. Our evaluated data do not reflect the high values of Batchelor and Wyld [14] at 2 and 3 MeV, although they are within the margin of error. Moreover, there is a significant discrepancy vis-a-vis the data of the English data library in the 3-6 MeV region and good agreement in this region with the data on the last evaluation of $\sigma_{nn'}$ in the German data library [20].

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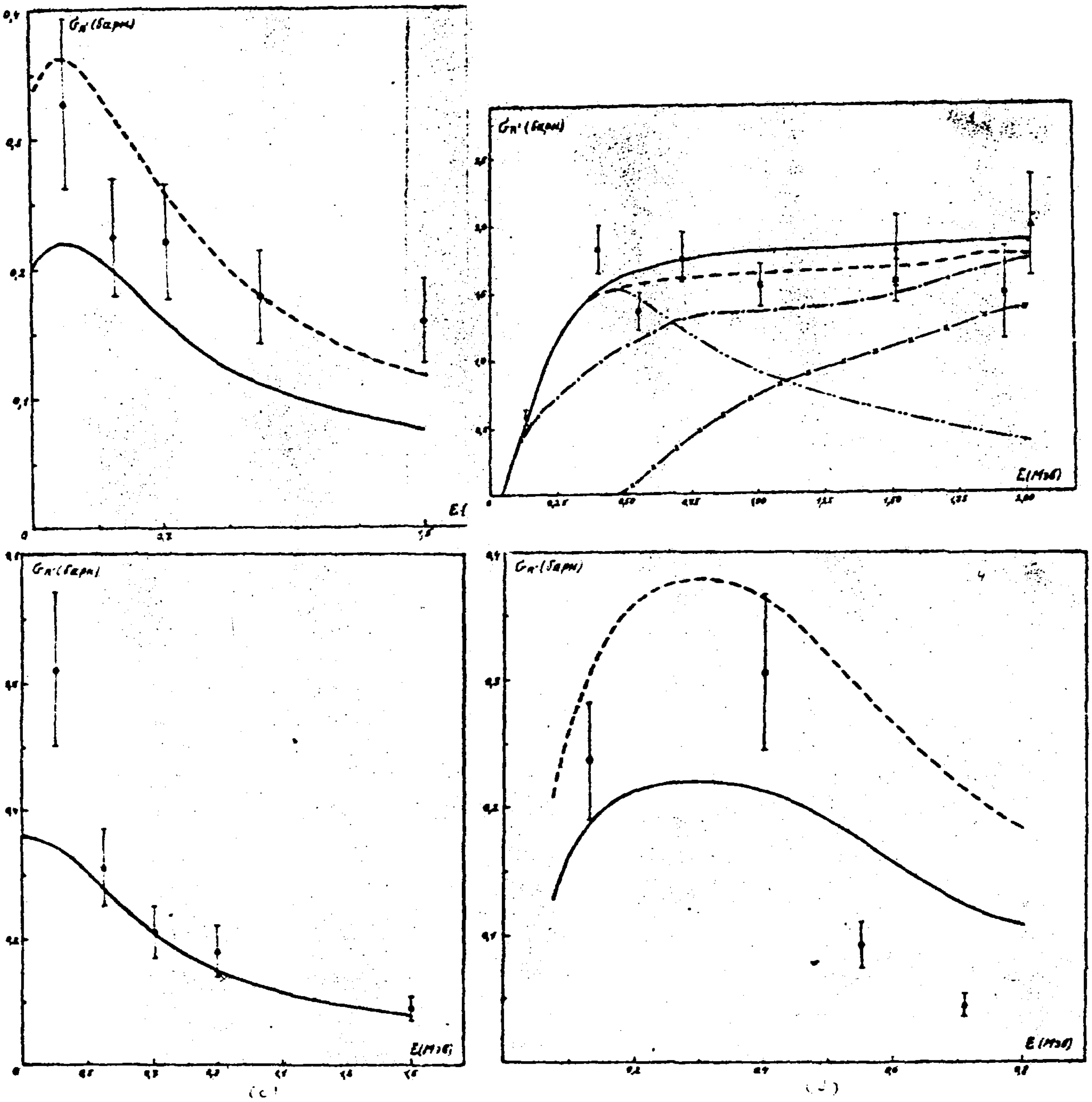

Fig. 1 Cross-section of inelastic neutron scattering by ^{235}U.

a) The excitation cross-section of the level group 100 < Q < 150 keV (. Armitage experiment; —— evaluation for the levels 103.0 and 129.26 keV; --- evaluation for the levels 103.0, 109.26 and 150.64 keV).

b) A comparison of the various data on ^{235}U in the region up to 2 MeV (- . - English data library; —— results of calculations for the present paper; ---- evaluated cross-sections $\sigma_{nn'}$; —''— resolved level excitation cross-section; ——x—— continuous excitation spectrum cross-section σ_{cont}).

c) Excitation cross-section of the group of levels 50<Q<100 keV (. Armitage et al., experiment; —— evaluation for the levels 51.73 and 81.63 keV).

d) Excitation cross-section of the group of levels 25<Q<50 keV (. experiment by Armitage et al.; —— evaluation for the level 46.16 keV; ---- evaluation for the levels 46.16 and 51.73 keV).