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TOTAL DELAYED NEUTRON YIELD FOR FISSION OF ^{235}U BY NEUTRONS OF ENERGY BETWEEN 10^{-5} eV AND 20 MeV. INCLUSION OF INTEGRAL INFORMATIONS

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__Abstract__ : The total delayed neutron yield from fission of ^{235}U has been evaluated using a model recently suggested by LENDEL et al which accounts for the incident energy dependence.

A scaling factor 1.0183 was needed to include the information from β effective measurements in clean cores.

The lower values summation calculations are well reproduced giving to the evaluated data additional reliability.

__Résumé__ : Le rendement total en neutrons de fission retardés pour ^{235}U a été évalué à l'aide d'un modèle récemment proposé par LENDEL et al qui exprime la dépendance en fonction de l'énergie incidente.

Un facteur de normalisation de 1.0183 a été nécessaire pour prendre en compte la mesure intégrale, dans des expériences propres, de la fraction effective des neutrons retardés.

Il en résulte un bon accord avec les valeurs basses de calculs de sommation. L'ensemble de ces convergences confère aux données évaluées un caractère de confiance.

1. INTRODUCTORY COMMENTS - THEORETICAL FORMALISM USED

As for ^{235}Pu [1], the total fission delayed neutron yield has been evaluated by making use of a recent phenomenological model proposed by LENDEL et al [2] which accounts for the energy dependence through an energy dependent explicit function $\varphi(A_f, Z_f)\ \Psi(E)$ and a terme $\omega(E)$ related to the prompt neutron yield $\nu_p(E)$.

The formula giving the delayed neutron yield y(E) at the energy E (in terms of a neutron number per hundred fissions) is written as :

$$y(E) = e^{a_0 - b_0 \omega(E)} + \varphi(A_f, Z_f) . \Psi(E) \qquad (1)$$

$$\varphi(A_f, Z_f) = -[2.626\ Z_f - A_f + 0.375(Z_f - 92) - 2.59] \times \frac{y_1\ (8\ \text{MeV})}{y_1\ ^{236}\text{U}\ (8\ \text{MeV})}$$

$$\Psi(E) = -\ a_1 + a_2\ E - a_3\ E^2 - a_4\ \text{th}\ (E - 5.5)$$

$$\omega(E) = \{0.1904 \times [2.626\ Z_f - A_f + \nu_p(E)]^2 + 0.1125\ (Z_f - 90)\}$$

$a_0 = 2.43$ $\quad b_0 = 0.725$ $\quad a_1 = 0.41$ $\quad a_2 = 9.68.10^2$
$a_3 = 2.13\ 10^{-3}$ $\quad a_4 = 0.25$

A_f and Z_f refer to the mass and the charge of the fissioning system.

The formula (1) directly results from the expression of the most probable charge Z_p of the fission product isobar distribution written as :

$$Z_p = \frac{Z_f}{2} + 0.71\ [N + \frac{\nu_p(E)}{2} - \frac{N_f}{2}]$$

$$+\ a\ \{5.98\ \frac{N_f}{Z_f} - 1.1\ \exp\ (-\ \frac{N + \nu_p(E)/2 - b}{3.64}) - 7.443\}$$

$$a = 1\ ;\ b = 59.4 \qquad \text{for } N \leq \frac{N_f - \nu_p(E)}{2}$$

$$a = -1 \; ; \; b = N_f - 59.4 \quad \text{for } N > \frac{N_f - \nu_p(E)}{2}$$

N stands for the number of neutrons in the fission product.

The coefficients have been obtained by least squares fitting on several nuclear data so that the $\Psi(E)$ function is identical for any fissioning nucleus.

A scaling factor is allowed provided it is compatible with the claimed systematic error of $\pm$ 7 %. In the case of ^{235}U target nucleus a small systematic error is expected as there is a normalization to the yield value of ^{235}U at 8 MeV of excitation energy.

A value of about 20 MeV (in the excitation energy scheme) is given as an upper limit of the validity domain.

2. EVALUATION

In the present work we imposed the constraints of consistency between :

- the integral measurement of β_{eff} in clean thermal spectra [3],
- the various summation calculations published in the open literature,
- the reliable data for ν_p and ν_t in the thermal range obtained by measurement or evaluation.

2.1 Integral data in the thermal range

There has been recent measurements of β_{eff} of ^{235}U in the thermal range at the Semi Homogeneous Experiment (SHE) facility at JAERI.

The advantages presented by the SHE facility are :

- Core composed of UO_2 and graphite whose nuclear constants can be considered as well known. The enrichment of the fuel in ^{235}U is 20 % so that the measured β_{eff} was the one of ^{235}U.

- The simplicity of the core geometry (cylinder) is adapted for accurate neutronic calculations, so that the obtained data can be considered as clean.

These experimental data are good candidates for benchmark testing.

Different configurations were used, characterized by different C to ^{235}U number of atoms ratios and therefore by different energy spectra. Consequently, the measured β_{eff} values were different ranging from 0.6802 to 0.6935 illustrating the sensitivity of this parameter to the in core energy spectrum. To note that the β_{eff} values were derived from different measured integral quantities such as ($\beta/\Lambda = \alpha$, ρ/β_{eff}, K_{eff}, $\Delta\alpha$) that have different sensitivities to the various parameter characterizing the delayed neutron emission (ν_{di}, λ_i, group i energy spectrum), the sensitivity being less for the last parameter [4].

All the conditions are fulfilled for a physically significant adjustment which has been performed with the least squares technique. In the adjustment procedure strong constraints were put on the group spectra. All the imput data were those of Keepin.

The conclusions by KANEKO and Coworkers [3] were :

- The adjusted β value should be 0.00677 $\pm$ 0.00008 and the KEEPIN's λ_i-values should be renormalized by a factor 1 - 0.004 under the assumption that the group energy spectra are correct. The error bar on β is surprisingly low (1.2 %). Most of the values indicated in the literature are at least twice as large.

- In a work to validate the nuclear constants in the thermal and epithermal range on a large number of integral data of different types (K_{eff}, reaction rates, spent fuel analyses) corresponding to a wide range of neutron spectra (different types of moderators and different moderating ratios), H. TELLIER [5] obtained consistent sets of adjusted data which are concerning ^{235}U :

$$\nu_t = 2.429 + 0.004 \ ; \ \sigma_f = 582.0 + 1.0 \text{ b} \ ; \ \sigma_\gamma = 98.4 + 1.5$$

2.2 Summation calculations

Summation calculations have been performed from measured/evaluated values for the cumulative yields and neutron emission probabilities of emitter nuclei, essentially in three different energy ranges, namely thermal, fission spectrum, 14 MeV.

The summation results may differ, sometimes significantly (~ 10 %) with the experimental [6], [7] or evaluated [8] data (restricting the references to the most recent ones).

The interpolations between ranges are generally linear and that is maybe questionnable. Sophisticated interpolation schemes, justified on a physics basis, are a real need.

2.3 Evaluations of ν_p and ν_t in the thermal range

Two exhaustive works aimed at a self consistent definition of the thermal constants for fissile nuclei together with the fission neutron yield from the spontaneous fission of ^{252}Cf have been published in a recent past. They are the works by DIVADEENAM and STEHN (1989) [9] and by AXTON (1985) [10]. Covariance techniques have been used in both works. For most parameters both data sets are consistent but that is hardly the case for ν_t of ^{235}U as it can be seen in the Table I.

A value $\nu^{th}_p = 2.415 \pm 0.0039$ is given by AXTON for the prompt neutron yield and has been adopted by FREHAUT [11] who showed the consistency of the data in the thermal range with the data at higher energies below 1 MeV.

Starting from this value for ν_p and from the integral value for β as derived by KANEKO and coworkers one obtains for the total delayed neutron yield ν_d ($\nu_d = \beta \times \nu_d/(1-\beta)$ a value 0.01646, with an associated uncertainty 0.00046 deduced from the accuracy on ν^{th}_p (0.2 %) and the one on β (1.2 %) and using a security factor 2. This value corresponds to a scaling by a factor 1.0183 of what is obtained with LENDEL's formula. It is perfectly consistent with the data given by COX [12], TUTTLE [8] or BESANT and coworkers [13] from experiments (a value $\nu_t = 2.4315 \pm 0.005$ for the total neutron yield follows from the hereabove data).

Summation calculations (REEDER [14], ENGLAND [15], MANEVICH [16]) result in higher values, leading to discrepancies concerning the β parameter which are unacceptable regarding the accuracy of the integral measurements (taking into account the security factor 2).

The couple of values $\nu_d = 0.01646$ and $\nu_t = 2.4315$ represents the consistency between microscopic and integral measurements and, concerning the total delayed neutron yield the consistency with a Physics based model.

The Table I shows some thermal constants for comparison purpose.

2.4 Evaluation in the other energy ranges

The evaluated values directly derive from the model using the same renormalization factor (1.0183) as in the key thermal range. First, second, and third chance fissions are taken into account according to their contributions to the total fission cross section, we have calculated using the statistical model, in a consistent way with JEF1.

$$\nu_D(E) = \alpha(E)\ y^5(E) + \alpha_2(E)\ y^4(E) + \alpha_3(E)\ y^3(E) + \dots$$

with :

$$\alpha_1(E) = \frac{\sigma_{f1}(E)}{\sigma_f(E)} \; ; \; \alpha_2(E) = \frac{\sigma_{n,n'f}(E)}{\sigma_f(E)} \; ; \; \alpha_3(E) = \frac{\sigma_{n,2nf}(E)}{\sigma_f(E)}$$

The numerical figures for $\alpha_1(E)$, $\alpha_2(E)$, $\alpha_3(E)$ are quoted in the Table II.

$y^5(E)$, $y^4(E)$, $y^3(E)$, ... are the delayed neutron yields corresponding to the first chance fission of the target nuclei ^{235}U, ^{234}U, ^{239}U, ^{233}U, ... respectively. Each delayed neutron yield has been calculated from the formula (1) using the corresponding prompt fission neutron yield, and normalized at the thermal energy on the value obtained from the systematics given by WALDO [18] :

$$y = \frac{1}{100} \exp\,(16.698 - 1.144\ Z_f + 0.377\ A_f)$$

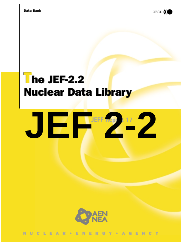
For ^{235}U the prompt fission yield results from our own evaluation between 10^{-5} eV and 120 eV and from FREHAUT's evaluation for higher energies, while the similar quantities for ^{234}U, ^{233}U have been obtained from the semi empirical formalism suggested by BOIS and FREHAUT [16] :

$$\nu_p^4(E) = 2.282 + 0.1399\ (E - 5.7) \qquad E \geq 5.7 \text{ MeV}$$

$$\nu_p^3(E) = 2.470 + 0.1634\ (E - 12.59) \quad E \geq 12.54 \text{ MeV}$$

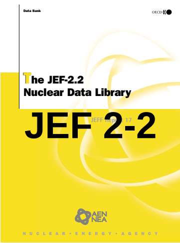
The energies are given in the system related to ^{235}U.

On the figures 1a and 1b are compared our evaluation with JEF1 and some chosen summation results.



The agreement is impressive from 10^{-5} eV to 7 MeV. At 14 MeV there is a 15 % disagreement with all the data including TUTTLE's one. Clearly there is a weakness of the model in the 14 MeV region which might involve WALDO's systematics especially for ^{234}U which has the major contribution to the total fission cross section. For this last nucleus, a thermal total delayed neutron yield value of 1.046×10^{-2} has been calculated.

Nevertheless the semi-empirical formalism by LENDEL and coworkers looks very promising for reactor applications.



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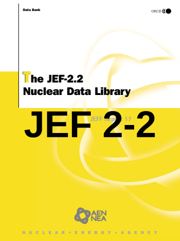

TABLE I - THERMAL CONSTANTS FOR ^{235}U

	β	ν_d	ν_t	Comments
KEEPIN [17]	0.0065	0.0158 ± 0.0005	2.430	Measurement
EVANS [18]		0.0163 ± 0.0013		"
COX [12]		0.0165 ± 0.001		"
BESANT [13]		0.0164 ± 0.0006		
TUTTLE [8]		0.01621	2.4188	Evaluation
AXTON [10]	0.00739	0.018	2.433 ± 0.0046	"
DIVADEENAM [9]	0.00684	0.01659 ± 0.00055	2.4251 ± 0.0034	"
ENDFBV	0.00685	0.0167	2.4367	
TELLIER [5]			2.429 ± 0.004	Adjustement on integral data
Present work	0.00677	0.01646 ± 0.00046	2.4315 ± 0.005	Evaluation
REEDER [14]	0.00764	0.0186 ± 0.001	2.4336 *	Summation
ENGLAND [15]	0.00727	0.0177 ± 0.0008	2.4327 *	"
MANEVICH [16]	0.00690	0.01678 ± 0.0009	2.4318 *	"

* Deduced from $\nu_p = 2.415$

TABLE II - RELATIVE CONTRIBUTIONS TO THE TOTAL FISSION CROSS SECTION OF ^{235}U

E (MeV)	First chance	Second chance	Third chance
0	1	0	0
1			
5			
5.7	1	0	0
6	0.9923	0.0077	
6.5	0.904	0.096	
7	0.739	0.261	
7.5	0.617	0.383	
8	0.562	0.438	
8.5	0.512	0.488	
9	0.475	0.525	
9.5	0.450	0.550	
10	0.424	0.576	
10.5	0.423	0.577	
11	0.421	0.579	
11.5	0.4175	0.5825	
12	0.414	0.586	
12.54	0.425	0.579	0
13	0.385	0.5547	0.06
13.5	0.365	0.528	0.107
14	0.345	0.515	0.140
14.5	0.326	0.503	0.171
15	0.311	0.496	0.192
15.5	0.303	0.489	0.208
16	0.299	0.476	0.225
17	0.285	0.956	0.259
18	0.270	0.436	0.299
20	0.260	0.420	0.32

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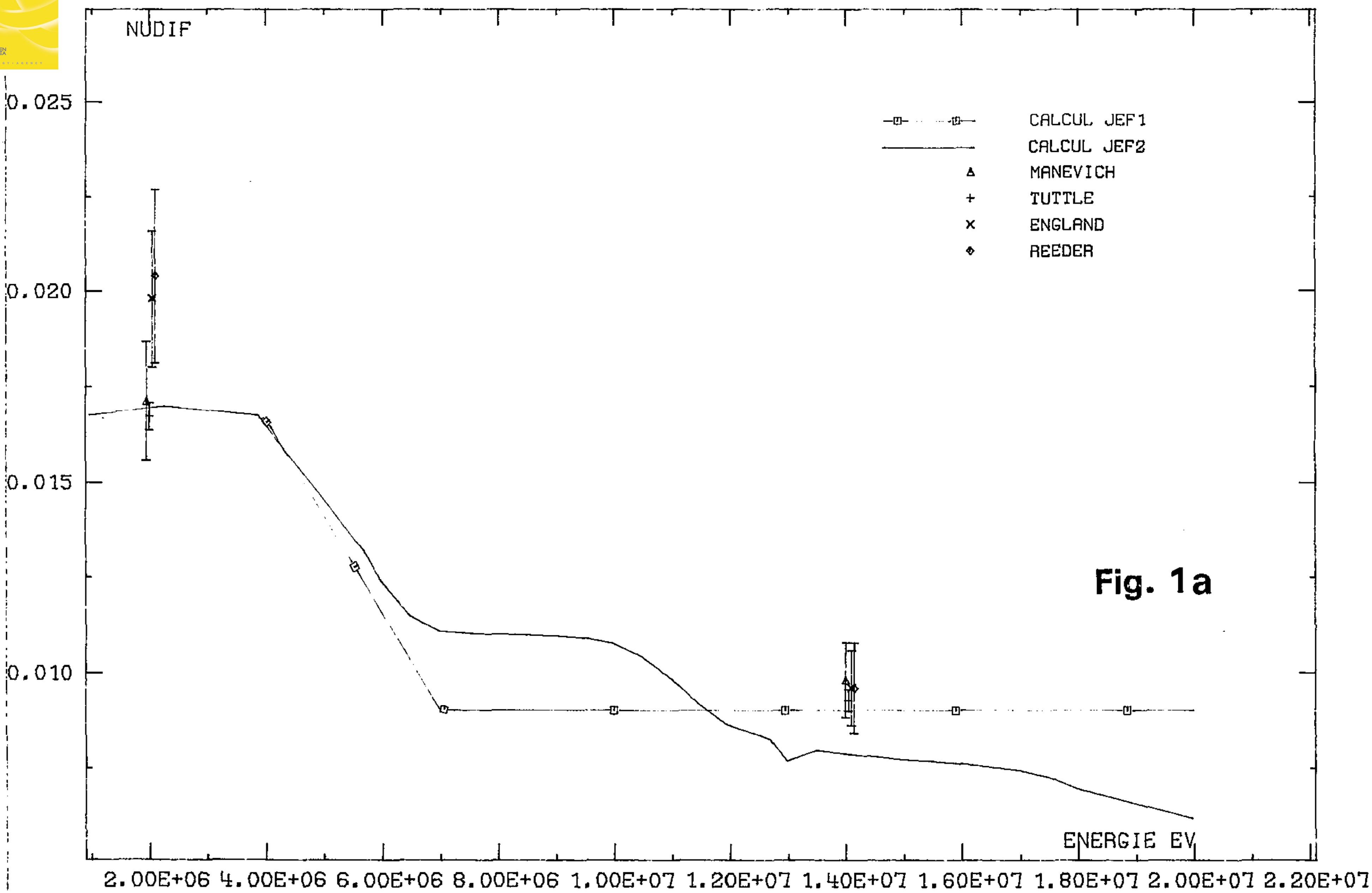

Fig. 1a

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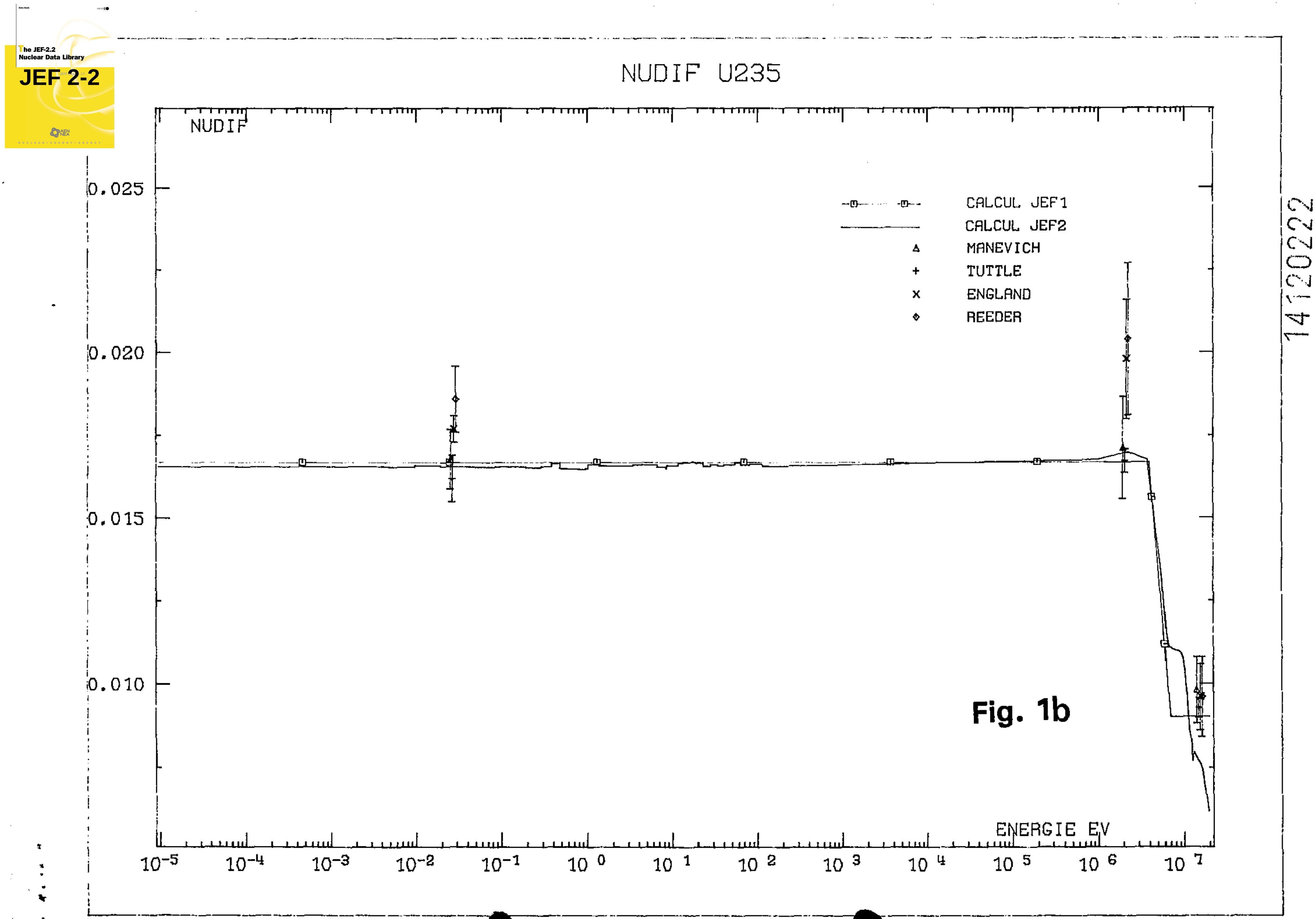

Fig. 1b

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